TRIP REPORT

United States LMFBR Steam Generator Team

IAEA Symposium, Bensberg,
Germany

October 14-17, 1974

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United States LMFBR Steam Generator Team

IAEA Symposium, Bensberg,
Germany

October 14-17, 1974

TNO, APELDOORN/STORK CO., Hengelo, Netherlands, October 18, 1974

INTERATOM, Bensberg, Germany, October 18, 1974

UKAEA/TNPG, Risley, England, October 21, 1974

EDF, Les Renardieres, France, October 22, 1974

EDF/STEIN INDUSTRIE, Paris, France, October 23, 1974

CEA/EDF, Marcoule, France, October 24, 1974

Acknowledgement

I wish to emphatically thank the team members for
their conscientious and professional response to
the meetings documented in this report.

Phillip A. Lowe

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I. INTRODUCTION

A. Purpose

The purpose of this team visit was twofold. First, it was desired for the United States to have a strong participation in the technical information exchange at the IAEA symposium on Steam Generators for LMFBR's. Secondly, the team desired to meet separately with the staff of several different European organizations in order to be able to explore more fully the details of some of the items presented at the general IAEA Symposium. To this end, individual requests for a U.S. team visit and a proposed agenda were forwarded to, and accepted by the specific organizations before the team initiated its travel.

B. Team Membership

Dr. Phillip A. Lowe, Team Leader

Chief, Steam Generator Branch

✓ Division of Reactor Research and Development

✓ United States Atomic Energy Commission

Mr. Robert W. Devlin, Manager

Steam Generator Systems Design

Westinghouse Electric Corporation

Mr. John S. McDonald, Program Manager

Heat Transfer Equipment

Atomics International

Mr. Scott R. Penfield, Supervisor

LMFBR Components

Combustion Engineer, Inc.

Mr. Wallace R. Simmons, Program Manager

Steam Generator Development

Argonne National Laboratory

Dr. Frank E. Tippets, Manager

Steam Generator Systems

General Electric Company

Mr. William B. Wolfe

USAEC Representative to Interatom

Bensberg, West Germany

Mr. Walter Wolowodiuk, Program Manager

Foster Wheeler Corporation

(Attended first week only)

C. Itinerary

The itinerary was as follows:

1. Full team participation at the IAEA Symposium, Bensberg, West Germany on October 14-17, 1974, including a trip to the Hengelo sodium test facility.
2. Messrs. Devlin, McDonald and Penfield to Apeldoorn, Netherlands (meeting with TNO and Nuclear Energy Project Group) and Stork Company at Hengelo, Netherlands on October 18, 1974.
3. Messrs. Lowe, Simmons, Tippetts, Wolfe, and Wolowodiuk to Interatom, Bensberg, West Germany on October 18, 1974.
4. The entire team with the UKAEA Fast Reactor Group and TNPG at Risley, Manchester, England on October 21, 1974.
5. The entire team with the French EDF at Les Renardières, Paris, France, on October 22, 1974.
6. The entire team with EdF and Stein Industrie at Stein Industrie, Paris, France on October 23, 1974.
7. The entire team with the French CEA and EdF at the Phenix power Station, Marcoule, France, on October 24, 1974.

D. Participants at Each Location

October 18 - INTERATOM - Bensberg, Germany

W. Ratzel, Manager, SNR Plant Component Design
Huebner, Sodium-Water Reaction Analysis and Relief System Safety
H. Mausbeck, Division Manager, SNR Development & Technology
D. Grosser, Manager, SNR Plant Materials
W. Jansing, Department Manager, Large Sodium Test Facilities

October 18 - TNO Nuclear Energy Project Group - Apeldoorn, Holland

A. R. Braun, Director, Nuclear Energy Project Group
M. L. G. van Gasselt, Manager, Sodium Testing
M. Hissink, Manager, Sodium Technology

October 18 - STORK Company - Hengelo, The Netherlands

N. Post, Engineer, Welding Department

October 21 - UKAEA Reactor Group - Risley, England

UKAEA

Jack Moore, Director for Fast Reactors
Keith G. Eickhoff, Head Sodium Technology and Components, REML
Jack A. Smedley, Engineering Technology Group, DERE
Ken Tregonning, Engineering Technology Group, DERE
Chas Robertson, Engineering Technology Group, DERE
Chris Tyzack, Materials and Corrosion, REML
Derek Harris, Overseas Liaison, Technical Operations
Directorate

TNPG

William Macrae, Director, Fast Reactor Projects
Derek Taylor, Manager, Fast Reactor Engineering
Owen Hayden, Manager, Steam Generator Engineering
Fréd Dickenson, NDE Inspection
Chris Willby, Steam Generator Engineering

October 22 - CGVS, EdF, Les Renardières, France

Henri Halle, Chief, Nuclear Technology Department
L. Lannou, Liquid Metal Division - Chatou
Llory, Liquid Metal Division - Chatou
Quinet, Liquid Metal Division - Chatou
J. P. Fontaine, Nuclear Technology Department
Jean Masse, Nuclear Technology Department
Saint-Paul, Nuclear Technology Department
Grand, Nuclear Technology Department
Madubost - EdF Dafeco - Paris
Vambenepe, Liquid Metal Division - Chatou
Ausange, Maintenance Supervisor, CGVS
M. G. Robin, Chief, Steam Generator Development, CEA Saclay

October 23 - Stein Industrie - Velizy - Villacoublay, France

Pierre Pouderoux, Director, Nuclear Department
Guy Salon, Responsible for Sodium Components
T. Nguyen-Thanh, Sodium Components
M. G. Robin, Chief, Steam Generator Development, CEA - Saclay

October 24 - PHENIX Reactor Plant, Marcoule, France

J. R. LeDuc, TECHNICATOME, Super PHENIX Project Manager
M. A. Valantin, EdF, PHENIX, Operations Supervisor
Henry Noel, TECHNICATOME - Saclay
M. G. Robin, Chief, Steam Generator Development, CEA - Saclay

II. SUMMARY

A. General

As an overview, the team found the discussions quite fruitful from several points of view. It is good to expose one's view-

points to the professional review of qualified but independent peers. The noting of differences in the various programs help highlight the strengths and weaknesses in one's own program. It was quite clear that all of the different national programs were supported by qualified, responsible individuals. It is the team's opinion that several viable programs are underway, and that the development of several concepts has advanced to the demonstration stage. It was the team's opinion that no foreign organization is clearly leading in the development of LMFBR steam generators (as compared to the US effort). There are, in fact, many similarities and overlaps between the several country's programs. The notable exception is the French effort epitomized by the Phenix plant. The French, more clearly than any other country, have demonstrated the ability to make the decision to build a demonstration plant and then get on with the job of doing it.

B. Steam Generator Design

1. General

This section summarizes the main design trends that appear to be evolving abroad for application in the period from now through about 1980, as evident from the papers and discussion at the IAEA Conference in Bensberg, (October 14-17, 1974), and from the subsequent US Team visits to Hengelo-Holland (October 16-17), Interatom-Germany (October 18), Risley-UK (October 21), Les Renardieres-France (October 22), Stein Industrie-France (October 23), and Phenix-France (October 24). The efforts on steam generator development in these countries are focused on either their respective "Demonstration Plant" or on the next plant to follow, depending on the current status of the Demonstration Plant. For perspective, the current status of the respective "Demonstration Plants" at the time of the team visits were as follows:

- . The 250 MWe PFR (UK) was operating at low thermal power (10 MWt) and operation is focused on the diagnosis of a small steam/water-to-sodium leak that appeared in No. 2 evaporator in early October.
- . The 300 MWe SNR-300 (Germany, Holland, Belgium, Luxembourg) is under construction on a schedule for start-up in 1976.
- . The 250 MWe PHENIX (France) has been in operation since March '74, and on the day of the team visit (10/24), the plant was operating at rated conditions delivering full power to the electrical grid. With this phase of LMFBR development apparently successfully accomplished, the French

have formed a consortium with Italy and Germany (France-51%, Italy-33%, Germany-16%) for engineering and construction of the 1200 MWE SUPER PHENIX.

- . The 250 MWe MONJU (Japan) is in an advanced stage of engineering, with the steam generator design selected (coiled-tube) and apparently is ready to proceed to the next phase of initiating manufacture and construction.
- . The BN-350 (Russia) has had steam/water-to-sodium leaks in three of its evaporators, beginning with a small leak in No. 2 evaporator in May 1973 (repaired and operation resumed) and including a large leak in No. 6 evaporator in September 1973 which actuated the pressure relief system and released 300-500 kg of steam/water into sodium; but despite this, at the time of the IAEA Conference (10/15/74) the BN-350 was said to be in operation on three steam generators, with the other three steam generators shut down for repairs. Apparently this experience has caused the Russians to move toward "modular" steam generators for future plants (presumably including BN-600, although not specifically stated at the meeting). The following items describe in summary the prevailing viewpoints and conclusions observed by the team regarding major aspects of steam generator design, particularly as they might be useful in guiding the US effort.

2. Types of System

The evidence noted by the 1972 US team visit to Europe (WASH-1219) of a trend in Western Europe to steam reheat and no sodium reheat is confirmed by the fact that the work on both SNR-300 (Ref. 50)* and SUPER PHENIX (Ref. 47) are proceeding on the basis of no sodium-heated reheaters. In contrast, the Japanese Demonstration Plant MONJU apparently is continuing to proceed on the basis of sodium reheat (Ref. 4). As is well known the British PFR uses sodium reheat (Ref. 1), and it was not learned whether or not sodium reheat will be used for the CFR.

The SUPER PHENIX apparently will definitely employ a once-through steam generator system with a major decision remaining to be made as to whether the steam generator will be of the "monotube" type (evaporator and superheater

*References refer to IAEA paper number presented in Appendix A.

in one shell without intervening manifolding) or will use separate evaporators and superheaters with nominally 30°C net superheat from the evaporator to ensure dry steam going to the superheater (Ref. 47). This decision apparently is related to the selection of the component design and vendor, with the choice to be made by June 1975 between two contenders: coiled-tube design by Fives Cail Babcock (combined 750 MWt evaporator and superheater in one shell) and straight-tube design by Stein Industrie (evaporator and superheater in separate shells, one of each in each 750 MWt loop). A 45 MW Babcock coiled-tube model has been tested at Les Renardieres; and a Stein straight-tube evaporator model and superheater model (45 MW total) are in the beginning of testing at Les Renardieres preparatory to the June 1975 decision between the two designs.

The SNR-300 uses a variation of the once-through approach which is actually a low-recirculation system (Refs. 50 and 24), and employs separate evaporators and superheaters (three evaporators and three superheaters in each of the three plant loops). In this system, at full power there is 95% quality steam exiting from the evaporator (5% moisture) with steam separation in a compact vertical steam drum and full liquid drain (5%) from the drum through the high pressure feedwater heater to the feedwater storage system. The method of level control in the steam drum includes automatic adjustment of the liquid drain rate from the drum. Full-flow demineralization treatment of the feedwater is provided in the SNR-300 system.

The MONJU plant uses a once-through steam generator system with separate evaporators and superheaters (and reheater) with control by exit superheat (30°C) from the evaporator (Ref. 4). Although the PFR uses a full recirculation system (5:1 recirculation ratio) and is said to avoid DNB except under certain transients (Ref. 1), the UK representatives indicated that for CFR consideration is being given to a "partial circulation" system, which is understood to mean high exit steam quality from the evaporator (similar to SNR-300), full drain of the separated liquid water back to the feedwater system and acceptance of DNB in the evaporator.

3. Evaporator and Superheater Configurations

The use of single-wall tubes appears to have been accepted in all countries as the reference approach for both the Demonstration Plants and the immediately subsequent commercial plants. As pointed out by the UK (Ref. 1), however, proof of the adequacy of the single wall approach remains to be obtained through operation of PFR (and PHENIX) over the next several months; and presumably a

rash of tube leakages in the operating plants could result in serious reconsideration of the multi-wall tube approach. In this latter regard, but apparently mainly motivated by the economic desire to eliminate the intermediate sodium loop if feasible, INTERATOM has completed a successful 3000 hour test of a "three-layer tube" (two Cr-Mo walls sandwiched with copper between them) of a steam generator which would use primary radioactive sodium directly as the heating fluid (Ref. 39).

There are three main single-wall design concepts currently under intensive development for use in near term plants that are to be built abroad, as follows:

- . Coiled-tube design: selected for MONJU (Ref. 4); selected for one loop of SNR-300 (Ref. 50); and a candidate for selection for SUPER PHENIX (Ref. 47). Figures 1, 2, and 3 illustrate the concepts for those three applications, respectively.
- . Straight-tube design with shell bellows: for two loops of SNR-300 (Ref. 50). Figure 4 illustrates the concept.
- . Straight-tube design with expansion bend in tubes: candidate for selection for SUPER PHENIX (Ref. 47). Figure 5 illustrates the concept.

The impression of the team, received from various discussions, is that the trend in the thinking appears to favor the straight-tube designs, rather than the coiled-tube designs, owing to their lesser complexity, improved capability for in-bore inspection, potentially lesser susceptibility to extensive damage propagation in the event of a large leak incident, and improved ability to extrapolate performance test results from comparatively smaller models to large plant units. Interpretation of this impression should be guarded, however, since the French state that the selection for SUPER PHENIX is yet to be made and will depend on the outcome of the tests of the Stein Industrie design now getting underway at Les Renardières (decision scheduled by June 1975); and for SNR-300 an immediate decision is side-stepped by the current plan of putting straight-tube units in two of the three loops and coiled tube units in the other loop.

The straight-tube design for SNR-300 has been successfully tested for 3000 hours in the Hengelo 50 MW facility, and the coiled-tube design is under test in Hengelo and at the time of the team visit had passed the 500-hour point. A similar test of the coiled-tube design candidate for SUPER PHENIX has been completed at 45 MW in Les Renardières and a

FIG. 1 COILED-TUBE DESIGN CANDIDATE FOR MONJU (ONE OF THREE CANDIDATE COILED-TUBE DESIGNS)

FIG. 2 COILED-TUBE DESIGN FOR SNR-300

FIG. 3 COILED-TUBE DESIGN CANDIDATE FOR SUPER PHENIX

FIG. 4 STRAIGHT-TUBE DESIGN FOR SNR-300 AND DETAIL OF INTERNAL BORE TUBE/TUBESHEET WELD

FIG. 5 STRAIGHT-TUBE DESIGN CANDIDATE FOR SUPER PHENIX

test of a 30 MW evaporator plus 15 MW superheater model of the straight-tube design for SUPER PHENIX is in startup at Les Renardieres. A test of the coiled-tube design for MONJU, including models of the evaporator, superheater, and reheater has been underway since May 1974 at the 50 MW Oarai facility in Japan.

It is noteworthy that none of the aforementioned designs for SNR-300 and SUPER PHENIX provide the feature of removable tube bundles. In contrast, tube bundle removability is provided for in the PFR steam generators and in the designs being developed for MONJU. The British representatives were particularly emphatic in stating the UK position that the removable tube bundle feature should be provided, in order to facilitate inspection and recovery after a sodium/water reaction incident to facilitate repair of leaks or modifications of the tube bundle--all more or less independent of whether or not cover gas is used in the steam generator unit. The SNR-300 steam generators will not have cover gas; but the straight-tube steam generator design for SUPER PHENIX does use cover gas (around the tube expansion bends), as do also the PFR steam generators and the design for the MONJU steam generators.

4. Materials Selection

The choice of materials for the sodium-steam/water boundary appears to be characterized abroad by a dominant concern for the problems of decarburization of unstabilized ferritic alloys such as 2 1/4Cr-1Mo, as chosen by the U.S.

It appears from the discussion at the conference that the UK is likely to remain with Nb-stabilized 2 1/4Cr-1Mo for the evaporator (as was used for PFR) and are considering 9 CR, 12CR, and I-800 alloys as potential candidates for the superheater.

For the SNR-300 the decision appears to have been firmly made to use Nb-stabilized 2 1/4Cr-1Mo for both the evaporator and superheater, and in fact the straight-tube models tested at Hengelo and the coiled-tube model now being tested at Hengelo all used this alloy for the tubing.

The most recent of the MONJU steam generators is proceeding on the basis of normal 2 1/4Cr-1Mo for the evaporator tubing and austenitic stainless steel for the superheater and reheater tubing.

The most recent design effort in France appears to be that of Stein Industrie for the straight-tube concept for SUPER PHENIX, and for this unstabilized 2 1/4Cr-1Mo (0.07-0.1%C) is used for the evaporator tubing and SS-316 is used for

the superheater tubing. The same material selection was used for the 30 MW straight-tube evaporator and 15 MW straight-tube superheater models built by Stein and now installed in the Les Renardières facility for testing. I-800 is being considered for the Babcock coiled-tube mono-shell design alternative for SUPER PHENIX.

From the foregoing, it appears that the U.S. is virtually alone in the world in its selection of normal 2 1/4Cr-1Mo for both the evaporators and superheaters, as has been done for the Clinch River Breeder Reactor Plant (CRBRP) steam generators.

5. Leak Detection

All of the steam generator system design efforts abroad assume the use of nickel diffusion-membrane hydrogen detectors, located in sodium taken from the discharge of each unit, as the primary means of early warning of steam/water-to-sodium leaks. Additionally, where cover gas is employed samples of the gas are taken for leak detection purposes.

It is generally stated but not operationally confirmed (excluding the Russian experience) that small leaks can progress in a short time to a situation of water release rate large enough to actuate the pressure relief system, either by self-damage or by wastage of adjacent tubes, and that in certain situations this time interval can be less than the time required to detect with hydrogen detectors and effect a shutdown. In view of this, the UK advocates automatic shutdown actuated by the hydrogen detectors, and in fact use such a system on PFR. In contrast, the French appear to not advocate automatic shutdown and on PHENIX, for example, depend on the operator to take appropriate action upon a series of progressing alarm signals of increasing hydrogen ramp rates displayed by the leak detector system. The actual UK experience with a leak is that they did not automatically shutdown upon detection, but operated for some time in order to confirm the existence of the leak and to study its propagation rate under actual operating conditions.

6. Sodium/Water Reaction Pressure Relief System

At the IAEA Conference there was considerable discussion of what the design basis large leak event should be, with the views ranging from one tube break being sufficient to the assumption that simultaneous guillotine failure of several tubes must be allowed for. At the end, the consensus was that the design criterion should be the equivalent of a single tube suffering a double-ended guillotine break as

the initiating event, but with the recognition that subsequently there may be additional tube ruptures due to rapid wastage or overheating which will add to the quantity and rate of water injection, which must be considered and allowed for in the design and sizing of the sodium/water reaction pressure relief system.

The SNR-300 will have rupture discs to the relief system located close to the sodium inlet and outlet of each unit, and it was emphasized by the INTERATOM representatives that the connection of both of the rupture discs to the plenums be as close-coupled as possible. Nominally, this system is designed to accommodate a double-ended guillotine of one tube, or the equivalent thereof; however, from discussions at INTERATOM on 10/8/74 it appears that sufficient margin is included to allow for the equivalent of a second guillotine break. The present design approach for the evaporators and superheaters for SUPER PHENIX indicates a sodium/water reaction pressure relief nozzle at the bottom of each unit. Additionally, it was learned from the discussions at Stein Industrie on 10/23/74 that an additional rupture disc relief is planned in this line in order to mitigate pressure pulses to the IHX. It is not clear how many tube ruptures the French assume as the design basis for SUPER PHENIX, but it was noted in the tour of the Les Renardieres facility on 10/24/74 that the relief pipe for each of the Stein straight-tube models (30 MW evaporator and 15 MW superheater) is at least 24 inches in diameter, which is consistent with that used for the CRBRP units. For PHENIX the statement was made that the relief system will accommodate the rupture of all seven tubes in a module.

The Japanese reported that for MONJU (Ref. 4) the design basis incident is the equivalent of four adjacent guillotine tube breaks which occur in the sequence of one tube failure followed by the failure of the additional three tubes. For intermediate size leaks there is a relief valve which enables bypass of the rupture disc to avoid bursting of the disc.

The designs of the sodium/water reaction relief systems are generally similar in that they provide direct relief to a sodium/water reaction products tank/separator and vent hydrogen to the atmosphere with provision for post-event throttling of the vent stack by a flapper valve of some sort and continuous purging of the system with inert gas. Neither the UK (PFR), the French (PHENIX and SUPER PHENIX), nor the INTERATOM people (SNR-300) believe that igniters on the stack to assure burning of the hydrogen are necessary--which is in contrast to the approach presently planned for CRBRP.

Apparently all the systems being designed abroad include on the steam/water side isolation valves, rapid blowdown provisions (20-30 second blowdown rate) and pressurization to about 150 psi with nitrogen--all operated automatically in the event of the main relief system activating--which is consistent with the approach presently planned for CRBRP.

The designs abroad also include isolation valves on the sodium side (about 10 second closure time) to provide further assurance of not getting reaction products into the IHX and to facilitate recovery operations--which is in contrast to the approach presently planned for CRBRP.

C. Development Programs

The West Germany (SNR), France (PHENIX), and Japan (MONJU) steam generator test programs include the testing of steam generator models of about 50 MWt capacities. In this size range, the cost of a test unit appears to be a weak function of unit capacity--if a test facility exists and is planned to be operated. The philosophy of these countries is to duplicate tube length, wall thickness, diameter, flow and fluid temperature conditions, and to construct as large a unit as is feasible to test. The British follow a similar philosophy except they use smaller test modules--few tube modules. In most cases transients selected are confined to the capabilities of existing facilities rather than trying to represent expected operational transients.

At the Bensberg Conference the French reported good agreement between the results from few tube model tests at Grand Quevilly, 50 MWt model tests at Les Renardières, and full unit performance in the PHENIX plant. A general concern was evident in most countries about small leaks and operator response to the indication of same. The USSR delegates were adamant that the steam generator should be shutdown immediately upon evidence of a small leak. Clearly, the USSR problems with BN-350 have had a strong influence. Both the British at PFR and the Dutch at Hengelo have experienced the difficulty of a small leak and the associated problems of locating the leak, i.e., operating the unit until the leak becomes large enough to find after shutdown, yet securing the system short of endangering the unit.

The development of leak detectors continues with emphasis upon total system response characteristics. Calibration of instruments is done both with hydrogen and with water injections. The French use hydrogen; the British and W. Germans favor water. The threshold of detectability is generally in the range of 0.05 to 0.1 g-water/sec.

The SNR-300 program has tested seal-welded, reverse-buckling rupture discs. The discs have an electron-beam seal weld between the disc and the pipe flange. Although the electron-beam seal weld is sensitive to the sulfur content in the base metal supports,

the performance of these discs are reported as very satisfactory over a 22,500 hour test period.

The possibility of corrosion on the water-side of the steam generator is a common concern, particularly in regard to water quality and the transition boiling region of the tube. The French have a fairly comprehensive program at Grand Quevilly to study these matters. No unusual corrosion problems were reported.

D. Fabrication and Inspection

Welding of the components which comprise the critical sodium-water pressure boundary (including attachments thereto) remains the principle source of steam generator failures to date. Three such failures were described during the conference. These involved a small leak in a PFR tube-to-tubesheet weld, a KNK unit failed at a weld point, and another occurred in the downcomer tube spacer in a Japanese test.

The integrity of the current weld design and welding processes are not yet fully proven by successful commercial operation. The degree of conservatism in various programs reflects this yet unsettled issue, as designers apply the results of their own experience. The Russians, worried by their misfortune with BN-350, are advocating modular construction. The British are considering thermal sleeves for CFR following the PFR leak. On the other hand, the French, buoyed by their still-perfect record, are designing large units for Super-Phenix.

A key problem remains in that nondestructive examination (NDE) techniques are not yet adequate to conclusively prove soundness of welds and materials when a no-failure criteria is necessary. In this case, however, considerable progress has been made.

The UK has demonstrated an eddy-current device which substantially overcomes the magnetic saturation problem found in ferritic steels. With this device they are able to detect 2 mil defects within the inner 20 mils of the tube wall. They also continue their development of small UT devices with considerable success.

X-ray techniques are being advanced within the SNR program. A new high-powered rod anode device is now under test at Henglo, Netherlands. They have overcome the magnetic beam deflection problem by a simple de-magnetizing device.

A completely different NDE procedure (for the US) was found to be in use in France. First a helium leak test indicates that a leak is present. The procedure then calls for pressurization with ammonia (NH_3) and detection of the leak sites with litmus paper. The French compare its sensitivity with the helium leak test.

Finally, in the NDE area, it was found that the British and French are using cleaning fluids as the hydrostatic test media. The British use "Genclean" while the French (Stein) use alcohol. The pressure test, therefore, also constitutes one of the final cleaning steps.

The ultimate justification of single wall designs may lie in the ability to detect and pinpoint the location of small leaks before they can grow. To this end the French and British are investigating acoustical detection along with correlative techniques.

While no unusual manufacturing techniques were revealed during the course of the visit, the European philosophy with regard to manufacturing standards was found to differ considerably from that in the U.S. In order to avoid needless expenditures, special standards for materials and manufacturing processes are developed only when commercial standards are determined to be inadequate. They concentrate their effort in a few critical areas such as tubesheet and tubing material, acceptance criteria for tubing and tube-to-tubesheet welds, etc. Additional standards, when found to be necessary, are developed jointly by the manufacturer and the purchaser.

E. Other Sites

The operations of the Phenix 250 MWe Demonstration Plant have been highly successful. After going critical in August 1973, the French moved quickly through the commissioning program and reached full power in March 1974. The plant has operated continuously since June 1974, except for a period of refueling which was recently completed. The performance of the SG's has been closely in line with predictions based on the performance of the 45 Mwt test model assembly at Les Renardieres and the earlier 5 Mwt test conducted at Gran Quevilly. The hydrogen leak detectors were calibrated by means of injecting hydrogen into the system, and apparently no anomalies have been observed during plant operations. During September, one secondary loop was shut down for the replacement of a butterfly valve in which leaks had occurred through two welds. The other two loops of the plant continued to operate during the repair period. After the valve had been replaced, the integrity of the new welds was verified by means of radiography and helium leak testing.

At the TNO laboratories in Apeldoorn, Netherlands, the single tube heat transfer tests being conducted in support of the SNR straight tube design will be concluded in October after completion of low flow stability studies. Testing of a single tube representation of the helical tube steam generator design for SNR will be initiated immediately thereafter. The helical tube model has already been installed in the test loop. The Apeldoorn tests have focused on heat transfer performance and stability, and no attention has been given waterside problems. The tube material for both the straight tube and helical tube model is SS 316.

III. TECHNICAL HIGHLIGHTS

A. IAEA Symposium

1. Summary

a. Design, Construction and Operating Experience of Past and Present LMFBR Steam Generators

The experience of several nations in the design, construction and operation of sodium-heated steam generators during the past 4 years has added a very sizable increment to the world's understanding of this key component of LMFBR power stations.

Large steam generator facilities of 30 to 50 MWt have been constructed and put into operation at Les Renardieres in France, at Hengelo, The Netherlands, at the LMEC in USA and in O-arai, Japan. Furthermore a number of other steam generator test facilities of up to 5 MWt have been operated in France, Germany, Japan and Italy.

Within the last two years we have witnessed the start-up of three large prototype LMFBR power plants, the BN-350 in the USSR, the 250-MWe PHENIX in France and the British 250-MWe PFR. Besides these power stations there continue to be useful increments of experience gained from the several steam-producing sodium-cooled test reactors, EBR-II, BOR-60, DFR and KNK, all rated at about 60 MWth.

So far, several large sodium-heated steam generators have operated for periods of 3000 to 5000 hours at power, and one unit, a PHENIX prototype of 5 MWth operated at Grand Quevilly for up to 8000 hours and a Russian pool type steam generator for up to 17,000 hours.

This experience has proven that the thermal-hydraulic calculational tools are suitable for extrapolating from the smaller test units to the larger power plant sizes. A number of general conclusions might be drawn from the papers presented by the participants and the subsequent discussions.

1. Steam generator configurations these days are not dictated so much by optimum heat transfer considerations as by optimum fabricability, quality assurance, inservice availability and detectability, and ability to detect and accommodate sodium-water leaks.
2. The concept of a single wall between secondary sodium and the water-steam side has become the nearly universally accepted approach.

3. Several different steam generator configurations such as bayonet, helical-coil, straight tubes, z-bend tube, hockey stick, U-tube and serpentine tube, all seem perfectly feasible and have not yet been shown to be significantly disparate in technical merit.
4. Once-through and up to fully recirculating steam generator systems are currently in-use with no clear patterns favoring any. Recirculating systems permit the use of lower water quality requirements, however, this may not be a very important consideration in view of modern water quality technology. Concern about low-load instability in once-through systems seems to be reduced by excellent performance at very low loads at several test installations.
5. Both liquid-filled and gas-blanketed steam generators are being designed, constructed and operated with good results. Some speakers presented experience that favored the liquid-filled system. These dealt with the problems of liquid level control in the gas-filled systems, and the greater assurance that the high pressure relief system in the liquid-filled steam generator will respond much faster and thus prevent structural damage to other tubes or to the shell.
6. The traditional concern about tube-to-tube sheet welds under sodium has been diminished by the excellent behavior experienced so far in several units. A certain number of leaks have occurred and they predominately occur in weld or in the heat affected region.

Recent developments in tube-to-tube sheet joints have shown several designs to be fabricable, inspectable and structurally acceptable. Thermal sleeves offer an attractive alternative for large power-rated steam generator units.

7. It is the consensus of designers to avoid stagnant sodium regions in the steam generator, primarily to facilitate early hydrogen detection in the event of small leaks.
8. The earlier notions of a "modular" steam generator are less clear as units become larger and larger with growing plant size. At this time there is no clear trend toward either a few large units or a large number of small units in LMFBR design practice.

b. Sodium Water Reactions

In the discussion of sodium water reactions the reports given showed that there has been continuing study in this field in all of the member countries along the lines laid down at the last Specialist Meeting. Now it can be concluded there are certain specific aspects which need emphasis and on which there was much common agreement.

These are:

1. Large water leak events

The incident of several water tubes failing simultaneously or within some milliseconds of each other, causing double-ended fracture of each tube was considered. It was agreed that such a concept was unrealistic, except for designs whose structure is such that failure of part of the structure could cause instantaneous mechanical damage and failure of more than one tube. Except for designs in this category, it was agreed that the design criterion should be a single tube suffering a double-ended fracture as the initiating event, even accounting for the escalation of small leaks (for example, the Hengelo reheater fault). It is recognized that following the initial event (double-ended tube fracture) there may be additional tube ruptures due to rapid wastage or overheating of these tubes, which will add to the quantity and rate of water injection, which must be considered and allowed for in the design and sizing of the sodium/water reaction pressure relief system.

2. Pressure relief systems to cater for large leaks

Pressure relief systems can be satisfactorily designed based on the results of tests already carried out in large facilities in the various countries of the world and now from experience in one reactor (BN 350). All systems to date have been based on rupture disc relief and the majority view was that for maximum effectiveness the discs should be located as close as possible to the component to be protected.

3. Small leak events

This was an area where much experimental work has been undertaken and many individual results have been obtained on wastage characteristics which, when considered overall, cover a range of materials under many operating conditions. However, at this time there is no single general evaluation of all of these results. Nevertheless it was recognized that although the resistance of Incoloy 800 is some 6-8 times better than Croloy at the same operating temperature and at the moment this covers the

range of resistance of the materials examined, this factor is probably not large enough to act as a predominant feature in the choice of materials for steam generators.

All of the work reported assumes that the leak is through a relatively circular orifice which gives a constant flow rate during the duration of the test. There was general concern that small leaks could suddenly increase in size either as a step change or with a very short time constant which would mean that the steam generator could not be shut down before further leaks would be produced. It has already been noted that the pressure relief system is able to deal with this eventuality. Here it is worthwhile reiterating the conclusion that from the available experience (Hengelo, Fermi, BN-350) the resulting set of multiple leaks will have had a combined flow rate less than that arising from a double ended fracture of a single water tube. However, there is a large economic incentive to stop the reaction before secondary leaks are produced because of the extended down time of the steam generator loop following such a major reaction. On this basis the majority opinion was that for any detectable leak the steam generator should be shut down; taking into account all the consequences of the difficulties of subsequent leak location.

However, if the leak can be found quickly then plugging of a single tube, or at the worst a few tubes could be a rapid operation which will allow the unit to be brought back into service with minimum loss of availability. However, it should be pointed out that it may be required to inspect and test the entire bundle before it is brought back to power.

4. Leak detection

During a sodium water reaction there are several physical or chemical phenomena which could be used as a basis for detection methods. However, at the conference, reports and discussion were all based on the production of hydrogen resulting from the reaction which was detected by diffusion through nickel membranes.

There were different methods of measuring the hydrogen coming from the membrane, but everyone reported satisfactory performance of their version of the hydrogen detection system. However, it was recognized that all of these systems have an intrinsic delay time between initiation of the leak and the first signal due to

hydrogen transport, etc., and therefore there is an incentive to develop a more rapid acting system (such as acoustical detection devices).

It was worth noting that some form of in situ calibration of the hydrogen detection system with water or steam was considered essential. It was recognized that hydrogen injection could only be used under certain conditions for calibration purposes.

c. Material Behaviour, Corrosion and Water Chemistry

A number of different materials have been considered for steam generator duties. Among these which cover 2.25 Cr-1Mo unstabilized and stabilized, Incoloy 800, 9 Cr-1Mo and 12 Cr-1Mo, respectively, the first named alloy has been given most of the interest. Despite the fact that this kind of material is known to decarburize on the sodium side, it has been used in a number of steam generators and is apparently favored in the US steam generator development programs. However, the preferred choice of this material is not recognized by all countries. The UK as well as the SNR-300 project have been more concerned with decarburization phenomena and have used the stabilized 2.25 Cr-1Mo steel. The other alloys - as mentioned before - have a number of attractive features as steam generator tube material; but there are some aspects which have not been completely clarified to enable their use in the near term steam generator application.

Concerning the sodium corrosion behavior of tube materials, the points of interest have been identified as the phenomena of mass transfer, including carbon transport and corrosion or deposition, respectively. As additional effects, fretting and wear between tubes and spacers were discussed but were classified not to be of major concern if adequate development work has been done.

Questions of metal loss on the sodium side are believed to be extensively investigated, but the effects of deposits due to corrosion product precipitations on the heat transfer performance of the steam generator tube need to be defined.

Problems related to the water-side corrosion of steam generator tubes have been identified by several authors. Especially, questions of the influence of high heat fluxes on the magnetite layer formation and as a factor which promotes the concentration of aggressive solutes were discussed. In addition to that, attention was paid to the corrosion effects possibly caused in the DNB region. As far as the experience is available from sodium-heated steam generators (which do not exceed a testing time with DNB of about 17,000 hrs steam time. The EBR-II steam generator has a steaming time in excess of 35,000 hours, but dryout, DNB,

is considered unlikely.) apparently no corrosion difficulties have been observed in the DNB region when operating with water of high quality; the anticipated service times of steam generator units, however, make it necessary to confirm this short-term experience.

For the operation of the steam generator, water quality specifications have been proposed which under normal operating conditions should not cause difficulties. Nevertheless, it seems to be worthwhile to perform some experiments which explore the corrosion and deposit behavior at the pessimistic side of the water quality tolerance band.

d. Maintenance and Repair

With respect to maintenance and repair there is the experience of defected tubes in some steam generator test facilities and power plants and of the dismantling of research steam generator units.

The question of maintenance was only touched. The amount of inspection of operating units could even influence the design of future steam generators.

Field repair in existing units proves to be a time consuming operation. Nevertheless there are proposals to use field repair in the future and to improve methods. The largest uncertainty in field repair is the time necessary to identify the defect tube in cases of a pinhole failure. There is a tendency towards replacement of the unit and repair in the workshop. Cleaning procedures have not yet come to a preferred solution. Methods like wet gas, alcohol and vacuum distillation are in use. Further experience will be helpful.

During the discussion it was stressed that careful in-service monitoring and internal inspection should be maintained.

There is a trend away from sodium reheat. It is recognized that there is a difficulty of field repair and this may be taken account of by the supply of spare parts.

e. Future Design

Even under consideration of the low statistical knowledge on steam generator failures there is a definite trend to larger units. This is dictated by the envisaged future plant sizes. This tendency might be limited by the detection capability for small leaks.

Many different configurations (for example, straight tube and helical types and operational modes, such as in once-through and recirculating steam generator systems) are probably technically sound. The key issues in steam generator reliability are the quality of fabrication and inspection.

2. First Day (Review of Presentations)

Hayden(1)* emphasized the need for testing of the actual plant hardware. This conclusion has resulted from start up experiences in which gas entrainment and sodium level fluctuations were encountered on PFR as a result of lack of understanding of the effects of design details and manufacturing tolerances on the hydraulic behavior at the sodium-gas interface.

On both DFR and PFR corrosion occurred when water found its way into steam generators which were layed-up for several months between installation and operation.

A baseline eddy current test of tubing as late as possible in fabrication was recommended as a result of experience on PFR in which corrosion occurred during pre-operational layup.

The British also reported that they used the following tube cleaning approach. An acid bath followed by felt parts being blown through the tube to remove rust. This was followed by examination and then plugging of some tubes.

Hayden(1) expressed a strong belief in the importance of removable tube bundles for diagnosis of design problems and maintenance.

The British(1) have developed an explosive welding technique and have used it to plug 40 tubes in PFR.

The Japanese(3) are utilizing a helical coil design steam generator with a removable tube bundle, which requires downcomer tubes from the top of the generator to the bottom where the bundle coils begin. Problems with excessive downcomer heating have led to adoption of a complex insulating system in which gas is used to displace sodium from within a baffled downcomer annulus. The resulting complexities tend to off-set the advantages of a removable bundle. The Japanese employ gas spaces in all modules. The sodium level is controlled by overflow from the evaporator and by cover-gas pressure in the superheater and reheater. The rupture disks are located in the gas spaces.

*Numbers indicate the IAEA paper number given in Appendix A.

The designers(5) of SNR specifically design without gas spaces in their components and place the rupture disk as close to the component as possible. The philosophy is to obtain quick rupture disk bursting thus limiting the extent of the sodium-water reaction by removing the sodium and thus avoiding the higher stresses seen by shells in components operated with cover gas. The initial spike early in the reaction without a cover gas while higher than the peak pressure encountered with cover gas, was found by experiment to be of insufficient duration to interact with the shell.

The SNR Na-H₂O reaction design basis is one tube rupture. This position was strongly defended on the basis of test results.

The SNR straight tube steam generator which employs a bellows in the shell for expansion has completed 3000 hours of testing and is currently undergoing examination. Early results show the bellows joint to have performed successfully. Tube examination after 1200 hours in the area of expected DNB revealed no physical indications. A slight discoloration was observed at tube support locations but no appreciable wear was detected. The SNR also employs thermal barriers in order to protect the tubesheets at either end of the module.

The SNR helical coil 50 MW prototype has been operated successfully for 500 hours at Hengelo. This model avoids the use of tubesheets by passing the tubes individually through the shell via thermal sleeves.

The Italians(7) are developing a straight tube steam generator without mechanical provision for expansion between shell and tubes. They have selected modules rather than large components as a result of an economical evaluation. The breakeven point was found to be two sodium/water reactions in the life of a plant. If there are more than two reactions, they feel modules are economically preferable.

The Italians(8) are developing two joints for their straight tube steam generator. The first employs a SNR type corner weld to a stub machined into the sodium side of the tubesheet. The second is a brazed joint in which the tube passes through the tubesheet and is brazed in a vacuum furnace. Both joints have been tested extensively and properties of the weld were found to be superior to the adjacent tubing.

The British(9) provided an extensive list of design criteria from the viewpoint of a utility customer.

The Japanese(10) design basis for sodium-water reactions is initial failure of one tube followed by the rupture of three additional tubes. This criteria is tentative pending the results of testing.

The Japanese sodium-water reaction effort is extensive in both analysis and testing. Of particular interest was a test in which water was injected into an array of 7 tubes of the MONJU geometry within sodium. The wastage of the tubes within the array was observed.

The Japanese have shown that a one dimensional sodium-water reaction code which employs a spherical model early in the reaction followed by a cylindrical piston model (S-C Model) will yield results slightly more conservative than a 2 dimensional model.

A seal welded, reverse buckling rupture disk was described(11) which has been tested to a life of 22,500 hours. While some problems with the electron beam seal welding have been encountered due to sensitivity to sulfur and phosphorus content, the operation for that period of time resulted in no detectable influence on rupture properties. Prospects are considered good for achieving rupture disks able to withstand a 10-20 year service life.

A Russian wastage test was described in which grades of material containing various chromium and nickel contents were exposed to a sodium-water reaction. The results tended to indicate a lower wastage rate with increase in either chrome or nickel. The test appears to compare a grade similar to T22 with a high chrome ferritic and with an anstenitic grade.

The Russians also reported that in large leak tests (5 kg/sec of water entering the sodium) for the BN-350 design, no secondary tube failure occurred. However, some adjacent tubes were bent or showed bulges in the reaction zone.

The Japanese (Hori)⁽¹⁰⁾ reported a calculational/experimental study the conclusion of which was that whenever a heat exchanger is modeled on the computer as a network of many parallel tubes, and if the program uses a one-dimensional pressure pulse calculational technique, then the calculated pressure in the heat exchanger is always greater than the experimentally measured value.

The Dutch (Ludwig)⁽¹¹⁾ reported experimental measurements that showed that the initial large pressure spike due to a sodium-water reaction in the steam generator does not produce a strain response in the unit. Thus, the pulse can be eliminated from design considerations.

3. Second Day (Review of Presentations)

Small-leak wastage test results(16) for 2-1/4 Cr-1 Mo for the two configurations of SNR-300 (straight-tube and coiled tube) were presented. Dumm introduced a jet theory approach for

predicting critical leak size and wastage pattern, and generalized to a wastage rate versus L/D with sodium velocity as parameter.

Metallurgical data were presented(17) on wastage zone, including microstructure, for small-leak wastage of 2-1/4 Cr-1 Mo and 18 Cr-10 Ni stainless steel.

An assessment of small-leak effects was reported(18), restricted to consideration only of adjacent tube wastage, providing estimate of typical times available for leak detection and shut down. Owing to short time available to respond to a leak situation, he suggested the possible need for automatically actuated shutdown. Also, he suggested (for PFR?) about 0.1 g/sec. as leak rate threshold for detection. Smedley discussed acoustical system development, and in discussion indicated that the current capabilities of detecting a 0.5 g/sec. leak rates with the present acoustic systems.

Povlanskii (discussion) stated that based on BN-350 experience, the Russian working assumption is that "small-leaks can become large-leaks in a few minutes, and that there should be automatic system shutdown.

The French(20) described the Phenix leak detection system, which includes multi-point sampling for each individual detector, and it described a method of computerized data correlation. In the discussion, it was stated that the French prefer hydrogen injection for leak detector calibration rather than water injection; and it was further stated that their test experience shows no difference between the two at sodium temperatures of 150-250°C.

Also, during the discussion (by German ?), it was stated that a small-leak in KNK was detected, then it appeared to plug. After shutdown and restart, the leak reappeared after 18-hours operation.

Smedley (discussion) stated that British strongly recommend water injection for leak detector calibration. Dumm (discussion) stated that water injection should be used for leak detector calibration.

A paper(21) described the development and performance testing of a nickel diffusion membrane leak detector, resulting in an estimated capability to detect leaks of 0.05 g/sec. in SNR-300. They believe that leaks smaller than 0.1 g/sec will cause negligible damage and will also not enlarge for more than an hour.

A paper(22) described leak detection requirements for SNR-300, which summarize to a ramp-rate of 0.3 ppm-hydrogen per hour.

The British(23) outlined tube failure experience involving "dryout" (DNB) for UK fossil-fired plants. From this experience, he concluded that the peak heat flux (for application to LMFBR steam generator) should be kept below 725 KW/m^2 ($235,000 \text{ BTU/hr-ft}^2$) to prevent accelerated corrosion. (Note: Bolt hardly mentioned any effect that improved water chemistry control might have to raise this proposed limit.) The fossils fired units are constant heat flux devices. Thus when fouling occurs, the tube wall temperature will raise so that a constant heat flux is maintained. This can lead to severe tube overheating at the dryout point. In contrast, a sodium heated device is a constant temperature device. Thus as fouling or dryout occurs, the tube wall temperature can increase to the sodium temperature, but as it does this the heat flux decreases. Thus the British criteria based upon fossil fire unit experience may not be applicable to sodium heated units.

Based on SNR-300 approach,(24) the paper recommends a feedwater pump driven recirculation system with 5% recirculation at full-load (95% exit steam quality), employing two feedwater pumps plus one spare per loop, with level control in the steam separator vessel controlled by adjustment of the 5% drain flow rate. He also recommended for the large leak design basis one double-ended guillotine tube break occurring 5 times in the plant lifetime; and for protection he also recommended steam/water blowdown to 15 bars and then inert gas pressurization of steam/water side after a sodium-water reaction event.

Robin (discussion) stated that Phenix and Super Phenix are designed for about 20% less than 725 KW/m^2 maximum heat flux. Robin also said that the French have conducted a single-tube test with a peak heat flux of 800 KW/m^2 for 5,000 hours steaming, and post-test examination revealed no evidence of accelerated corrosion at the DNB point. (Note: In private discussion (Robin/Tippets), Robin said these tests are continuing and he expects to reach 10,000 hours steaming at the high heat flux by the end of CY-75).

Ratzel (in discussion with Tippets/Devlin) stated that the principle reason he recommends 95% exit steam quality from the evaporator (a la SNR-300), rather than, for example, 20°C net superheat, is to avoid excessive temperature cycling of the evaporator exit tube sheet and the superheater inlet tube sheet.

The rationale(25) for UK materials selection, stating the UK position that (in contrast to US view) 2-1/4 Cr-1 Mo decarburizes excessively was discussed; and after consideration of I-800 and stabilized 2-1/4 Cr-1 Mo as alternatives, the British appear to be trending toward selection of 9 Cr-Mo for CFR.

The results(27) of detailed experimental studies of deposition and the characteristics of the magnetite layer was given, and corrosion rates of 2-1/4 Cr-1 Mo, I-800 and I-625 in super-heated steam with heat flux. In the discussion, Grosser stated that he has tested a once-through steam generator (about 5 years ago) and determined that there was no disturbance of the magnetite layer at the DNB point.

A list(30) of general criteria as a guide for the preparation of maintenance and repair procedures for SNR-300 was given. Acid-cleaning presently is not being considered for the SNR-300. The results(30) of experiences in using a vacuum distillation method for cleaning the sodium-side of a full-scale (50MW prototype) steam generator were also given. For this, the unit was kept at 400°C using trace heater and hot gas circulation through tubes. Good results was obtained; however, the vacuum equipment needs further work.

The French(32) presented test results from the 7-tube model of the Phenix steam generator, and compared the model test results and analytical predictions (including preference with plugged tubes) against recent measurements of actual Phenix operation. There is apparently good agreement between the model tests, analytical predictions and the Phenix operation.

The Russians (Sokolov) gave a description of recent problems in the operation of BN-350, including in particular the incidents of small and large leaks that have occurred, beginning in May 1973. Three of the evaporators have had leaks. Three steam generator loops are now in operation, and the other three are down while the steam generators are being repaired.

In May 1973, a small-leak occurred in Evaporator No. 2 during low power operation, starting at a rate of 8-10 lb/hr, eventually growing larger, from which a total of 3-4 kg of water was injected. After shutdown a 0.2-0.3 mm hole was found at the end-cap weld (bayonet tube geometry). Repairs were made (tube plugging) and currently about 9000 hours of operation have been accumulated since the leak (at loads not exceeding 55%). The leaky tube was found by pressurizing the tube side with helium.

In September 1973, after 3000 hours operation, a large leak occurred in Evaporator No. 6. There was an initial indication from the leak detector of hydrogen concentration increasing in the cover gas space; and a few minutes later the shell side pressure (measured in the cover gas) increased rapidly to 2.5 kg/cm², at which time the rupture discs burst, and the sodium was dumped. During the period of shell side pressure increase a steam/water blowdown was initiated by operator action. A total of 300-500 kg of water is estimated to have leaked into the sodium. It was found during examination that

a tube had cracked. The dumping system for the steam generators worked as intended; and it was also stated that the reactor was not scrammed (!) In the discussion of the paper Sokolov said that the steam/water dump was initiated in the 5th minute after the event started and took about 5 minutes to complete so that at the 10th minute the steam/water pressure was reduced to 1-2 bars.

During the discussion of the Sokolov paper, Povlavskii (USSR) confirmed that owing to the difficulties experienced with BN-350, the USSR is presently tending toward modular steam generator designs. Apparently, they are hoping to be able with the modular design to automatically cause rapid shutdown of an individual module upon detection of a leak in the module, and, thereby, avoid having to shutdown the entire steam generator loop.

The detailed measurements⁽³³⁾ of tube wall temperature oscillation in the DNB, transition boiling and film boiling region with a sodium-heated tube were presented. These tests indicated, among other things, that the transition boiling region is short (about 10 cm long in the test).

4. Third Day (Review of Presentations)

The AI MSG(34) test program was reviewed and the principal design features of the model steam generator were described. It was noted that the MSG obtained 9000 hr of sodium operating time during which the unit was steaming for 4000 hr. Two 500-hr endurance runs were made; one in the once-through operational mode, the other in the evaporator mode of operation. Three areas were discussed in detail; thermal performance, leak detection arrangement, and the results of the metallurgical examination of representative tubes. The latter showed no deleterious results.

In response to discussion from the floor the following information was developed. Orifices are provided at the inlet to the tubes of the evaporator. The pressure drop across the orifice is nearly equal to the remaining pressure drop along the tube. Performance of the unit was in good agreement with predictions. Differences of note were 1) DNB occurred at twice the quality predicted, 2) film boiling coefficients were different than prediction and 3) the unit pressure drop on steam side was slightly over predicted. At design sodium conditions the observed power was 32 MWt as compared to a predicted value of 28.4 MWt. No deleterious wear occurred to the tubes at tube-support regions. McDonald reported, as did Sano (Ref. 35), a black powdery coating on the sodium-side after cleaning.

A review⁽³⁵⁾ was presented of the operating experience with the PNC 1 MWt, once-through, helical-coil steam generator and the related test program. The unit was completed in 1971, and various performance tests were carried out in the

1971-1972 period. Excessive heat loss was encountered in the feedwater downcomer tubes. The unit was dismantled for inspection and improvements. It was restored to operation and incurred two small leaks. The small leaks occurred in the downcomer tubes which had been modified to correct the heat loss problem. Limit cycle instabilities were developed by throttling the inlet flow. The unit was cleaned by drawing a vacuum at temperature and received a final steam cleaning after bundle removal. Black powder was observed on the sodium side of the tubes after cleaning.

The operating experience with the 50-MWt "MONJU" model steam generator was described.(36) This unit is currently under test. The unit reached full power in June 1974 and has achieved 1500 operating hours. Difficulties with the control of the liquid levels in the separate evaporator and superheater units were reported. Plans are to modify the unit to remove excess heat transfer surface area and to improve control of the liquid levels.

During the period April 1972 to June 1974 a 50 MWt straight tube steam generator was tested in the TNO 50 MWt SCTF at Hengelo.(37) The unit was tested for 3000 hr. A detailed description of the construction aspects of the test model, thermal and hydraulic performance, and various operational aspects are contained in the paper. It was reported that the test unit had a large bypass flow between the tube bundle and the shell. Instability in performance of the outer tubes was reported as being similar to that which occurred in the BLH/ALCO steam generator. The instability was described as a sudden shift to a lower operating point. Increasing sodium flow, all other things held constant, caused a sudden drop in feedwater flow and power. It was postulated that too large a bypass flow may be the root of the problem.

Operation of a 1 MWt, helical-coil test section was reported representatives of CNEN.(38) The tests were described as satisfactory.

The testing of a bayonet type, multi-wall (3) tube was reported.(39) The triple-walled tube contains leak detection ducts between each wall similar to the Westinghouse duplex tube. The tube was tested for 3000 hr. Although small gaps developed between the tubes, it was reported that no change in wall conductivity was observed. Leak detection operated satisfactory on the water side but was unsatisfactory on the sodium side.

The 45-MWt helical-coil unit in test at Les Renardaires, France has a lower pressure drop than predicted.

The US CRBRP water-side Chemistry program was described.(40)

Water-side corrosion was investigated in steam generator and test units in the W. Germany SNR program. It was reported:⁽⁴¹⁾

1. a sodium-heated steam generator is more reliable than a fossil-fired unit when water-side is being considered,
2. operation with neutral water is satisfactory thereby permitting chemical analysis, after the startup period, to be reduced to random tests,
3. high pressure drop due to rippled magnetite has never been observed, and
4. magnetite layers remained stable under all loads including severe transients.

Hydrogen concentration as a function of load change and operating time was also investigated.

During the discussion the W. Germans reported satisfactory performance with their condensers and feedwater heaters. Materials used are 15M03 and 13 Cr. M044. The designs included features to reduce fluid velocities in the tubes and use of a special nozzle at the tube plate to smooth the flow and reduce turbulence.

The French⁽⁴⁷⁾ have made a major change in their steam generator design philosophy in moving from small modular units (e.g. Phenix) to very large units. As a result they showed great interest in leak detection and location, automatic or manual response, and blowdown procedures and relief time.

Tour of Hengelo

The 50-MWt Sodium Component Test Facility (SCTF) at Hengelo, Netherlands is operated by the Central Technical Institute, TNO. The group of about 56 people at Hengelo is involved in the following areas:

<u>Component or Item</u>	<u>Areas of Work</u>
Steam Generators	heat transfer fluid flow - distribution - vibration thermal shock special calculations - stress - vibration
IHX	fluid flow heat transfer (full-scale tests)
Pumps	shaft vibration calculations bearing materials bearing development visco seal development

<u>Component or Item</u>	<u>Areas of Work</u>
Materials	mechanical properties at high temperature corrosion and deposition
Welding Research	repair techniques of sodium-contaminated materials
General Support	cold traps, valves, chemical analysis of Na, H ₂ detector development

Maximum thermal power of the test facility is 57 MWt; however, IHX's can be tested to 70 MWt. Feedwater temperatures of 300°C are obtainable. The unit is operated with 4 operators per shift.

The following items of interest were observed during the tour.

- . Na spark-plug-type leak detectors are located only on valves
- . Heatup rate of the facility is 5°C/min
- . Analog and digital data logging and operational control
- . Valve maintenance is performed yearly; instituted as a result of frequent valve problems encountered during commissioning
- . Four patented mixing tee designs are being tested--the tees can be purchased
- . The IHX is removed from service and the bundle was removed from the shell. The bundle was first drained (by a tube from the top) and pulled into a plastic bag. About 20 to 30 liters of Na was left in the bottom of the tube bundle
- . A number of Na leaks through the seal welds of the upper flanges of the IHX occurred early in the test program
- . The sodium system is designed to drain automatically if the rupture discs burst

5. Fourth Day (Review of Presentations)

On the last day of the conference, papers were presented in the areas of "Steam Generator Faults" and "Future Design Aspects," and final conclusions of the meeting were drafted. In addition, a detailed report was presented regarding the detection of a leak in the PFR No. 2 steam generator. Highlights of information presented in these areas are reviewed below.

Steam Generator Faults

The two leakage incidents experienced with the Japanese 1 Mw steam generator helical tube model(42) were found to be attributable to manufacturing problems. The leaks occurred in the heat affected weld zone of a 2-1/4 Cr-1 Mo downcomer tube-thermal insulation pipe assembly. It was concluded that the problem was the result of omitting the post weld heat treatment of these welds, and - on the basis of this experience - it was decided that the 50 Mwt model would have butt welded joints in this region which would receive post weld heat treatment.

A tube leak occurred in the KNK steam generator in September 1972.(43) The leak site was in one of the spacer welds which joined tabs to the steam tube. These tabs maintain clearance between the water/steam tube and the larger, concentric sodium tube. The outer faces of the fins are Stellite with 10 chrome Mo, Ni, Nb 9-10 steel. The fins were joined to the tubes by a TIG weld. A thorough examination of the failure site as well as conditions at 18 additional tab welds in the assembly led to the conclusion that the leak had been the result of a weld "tail" which escaped detection during the manufacturing examination. No "tails" were observed in any of the other fins examined. Steam generator operations have continued since the occurrence of this problem, and no other difficulties have been encountered after a total of 5200 hours of operation.

Leakage indications occurred during operations of the Hengelo test facility in August 1972 (when the system was undergoing operational checks) and in April 1973 (during the course of a dynamic test). The first leak was confirmed and was determined to have been caused by a tube to tubesheet weld in which there was a lack of penetration. The leak occurred in the reheater module and resulted in a gas bubble which almost completely blocked off the sodium flow through the unit. Eventually the bubble was pushed out the exit of the module. This experience resulted in a revision of the shutdown procedure, as described in the paper by C. Smit.(45) Investigations following the April 1972 leak signal led to the conclusion that the indication was false. In 1800 hours of additional operation, there has been no repetition of such a signal.

The Japanese indicated that in examining the Fermi No. 1 steam generator, numerous pits have been discovered in the lower section of horizontal passes of the serpentine tube bundle. One pit was completely through the tube wall and a number of others had penetrated almost all of the way through. The Japanese were unsure as to what the cause might have been, however, the pitting was characteristic of oxygen damage during the layup period. The UK reported

on the status findings regarding a leak recently detected in the evaporator of the No. 2 steam generator. The PFR steam generator units had originally been subjected to helium leak tests which assured leak tightness down to at least 10^{-6} g/sec equivalent water. This corresponds to the helium leak test detection sensitivity. After steam generators had been installed in the plant, but before they were filled, an additional helium leak test was conducted with the loop at a temperature of 400°C. This condition was held for several days after which there occurred an increase in the evaporator cover gas pressure. A short time later the hydrogen indication diminished to the original value, and the operators concluded that possibly a leak had opened and subsequently closed. It was at this point difficult to be certain whether a leak had existed or not. It was determined that commissioning activities should continue.

In late September, the secondary system was filled with sodium and water, and no anomalies were observed. Later a second helium leak test was conducted with sodium in the units at approximately 400°C. The test was conducted with a helium-argon mixture pressurizing the tube side at 500 psi. For several days the helium content in the cover gas remained at background level. Then for a period of time the level rose well above background (helium indications increased by about a factor of 25). The magnitude of the apparent leak was estimated to be of the order of 5×10^{-5} g/sec of water. Subsequently, the tube side was subjected to a series of pressure cycles, however, the effort to reproduce the leak was unsuccessful. Steam was then added at a pressure of 90 bars, and later - as there was no further evidence of a leak - water at 117 bars was added. The hydrogen detection system gave no indication of leak for a period of about one week, and it was then tied into the automatic dump system. Shortly thereafter, a dump of the water system occurred which was triggered by a "vibration sensitive" cover gas hydrogen indicator which was disturbed by a workman walking near the instrument.

The steam generator was recharged with water and, after a period of normal conditions, a new indication of a leak developed in the cover gas region near the sodium inlet. The hydrogen signals corresponded to a water leak rate of 2×10^{-3} g/sec. Operations were continued under this condition for 90 minutes (the leak indicators had been disconnected from the automatic trip system at this time), and then the system was dumped. After a further investigation, low pressure steam was introduced to the tube side and was gradually raised to a level of 117 bars. When a pressure level of 117 bars was reached, the leak indication

reappeared. This signal corresponded to a leak rate of 8×10^{-3} g/sec. After holding at this condition for an additional 90 minutes, the decision was made to shut down. It was concluded from the evidence available that the leak had occurred in the region of the tube plate above the sodium level. In order to study the leak characteristics for a longer period of time, an attempt was made to evacuate the cover gas and flood the shell side with sodium up against the tube plate. This would enhance the detectability of hydrogen through the use of the in-sodium detectors. The tube side of the unit was filled with water. It was found that as soon as the tubes were filled with water, hydrogen indications were signaled in the gas space and by the inlet region sodium level indications. The system was dumped again, and it was estimated that the final magnitude of the leak was in the range 0.1 to 0.25 g/sec. At the time the above report was presented, the tube side had been refilled with steam at 50 bars, and the leak indication had subsided to a level of the order of 10^{-4} to 10^{-5} g/sec.

It has been decided that the No. 2 unit will be shut down, the head removed, the leak located, and the faulty tube or tubes plugged. In attempting to explain the alternating appearance and disappearance of the leak indication, it was suggested that a tube-to-tubesheet weld had opened up. It was postulated that, with the addition of steam, magnitite was forced into a position where the hole would be blocked. Then, when flowing water was added, magnitite was removed and the hole opened up. During the history of the leak, no step changes in hole size occurred. Rather, the opening appeared to be gradually increasing in size. An attempt will be made to accomplish the repairs indicated above while sodium remains on the shell side. The approach to be followed in searching out the faulted tube will be to plug one end of the tube while the opposite end is pressurized with gas including hydrogen. Tubes will be pressurized one at a time in the search, and the gas face will be sniffed for evidence of hydrogen. The possible use of listening devices is also under consideration.

On the basis of the above experience, the British have concluded that automatic shutdown in the event of a leak indication is essential. They also believe that the number of spurious trips which might be caused by instrumentation will be minimal. They indicated that their hydrogen leak detectors are now all tied into the automatic trip mode. It was generally agreed that the set points used on such hydrogen detection equipment must take into account (1) the time in which a small leak can result in failure of adjacent tubes, (2) the desire that small leaks remain open as long as possible to facilitate location and correction. Based on the small leak evidence

observed to date, the British believe that the occurrence of a step increase in the size of a small leak is unlikely.

Future Design Aspects

Descriptions of three steam generator designs being considered for Super Phenix are provided in the papers⁽⁴⁷⁾ presented at the conference. These units are: (1) the Five Cail BW unit, (2) the Stein units, and (3) the Italian straight tube modules. Tests of the five tube 48 Mwt Five Cail unit were completed at Les Renardieres in December 1973, and performance results were generally in line with predictions. Tube side pressure drops were found to be slightly lower than predicted, however, because of an overestimate of the tube roughness. Tests of a three tube model of the Stein design have already been concluded as has a hydraulic model test. Results indicated that the steam quality at the departure from nucleate boiling was higher than predicted using published correlations. A 45 Mwt assembly of 102 tube evaporator and 37 tube superheater models of the Stein design will undergo testing in Les Renardieres starting early in 1975. Tests of a 1 Mwt model of the straight tube Italian design⁽⁴⁸⁾ have been completed, and preparations are underway for the long-term testing of a 50 Mwt model. The French indicated that in May-June 1975, a steam generator design would be selected for use in Super Phenix.

A study was presented by the Germans⁽⁴⁹⁾ at the IAEA conference which indicated that the trend in future plants will probably be toward modularity in the interest of (1) enhancing leak detectability, and (2) facilitating early recovery. However, it was recognized that this approach will incur increased capital costs compared with the use of either single large units or a few large units in each loop. As a possible solution to the problem of trading off leak detectability-recovery against capital costs, a pool-type unit incorporating replaceable modules was suggested. Another alternative suggested⁽⁵⁰⁾ was the use of an integrated three fluid steam generator unit which coupled the primary system sodium directly to the steam system.

B. Interatom Meeting October 18, 1974

The SNR-300 cost estimate is 1.7×10^9 DM which includes 0.5×10^9 DM for escalation. It is scheduled to go critical in '78 and be turned over to the utility in '79.

The program for the development of the NSR-300 tube-to-tubesheet weld took several years to complete and over 300 weld specimens in order to qualify the procedure. The weld looks like:

It is a one pass fusion weld with no feed.

In the SNR-300 program the tubes are delivered inside of plastic sheets. There is no requirement as to the material content of the plastic. There are no special welding floor cleanliness requirement above that which is standard for the Dutch industry (i.e., separating austenitic from ferritic materials, etc.). The ASME code is used for calculations, but not for welding or assembly. In those areas a West German code is used.

A final hydrostatic test (with water) for both sides of the steam generator and also a helium leak test are required.

The SNR-300 prototype test included:

- a. Endurance tests at full load for about 3000 hours
- b. Application of plant transients (for example load ranges of 20-100% at 10%/minute)
- c. Low (decay heat removal) operation.

Items b and c consumed about 5 hours of test time.

The Interatom test program is based upon performing full load model tests to define operational characteristics. Their test requirements do not define scaling criteria between the test model and the plant units. The model size is based upon:

- a. The maximum power level which can be tested in available test loops.
- b. Duplication of the plant unit tube length, diameter and wall thickness.
- c. Matching the model inlet and outlet temperatures to the plant-unit values.
- d. Matching the plant and model mass velocities ($\text{kg/m}^2\text{sec}$), i.e., same power/tube as in plant unit.

- e. Carefully measure all flow distributions.
- f. Perform steady state performance tests.
- g. Apply plant transients to the extent permitted by the loop capacities.
- h. Destructively examine the model after the test program is complete.

The above modeling criteria has lead to the development of Table I.

Table I

German Model Test to Plant Limit Comparisons

<u>Item</u>	<u>Hengelo Model</u>		<u>SNR-300 Limit</u>	
	<u>evaporator</u>	<u>superheater</u>	<u>evaporator</u>	<u>superheater</u>
Power, MW	28	17	55.4	30.1
Number of tubes	139	139	211	211
Length (tube sheet to tubesheet), m	19	14	20.0	15.3
Tube OD, mm	17.2	17.2	17.2	17.2
Tube wall thickness, mm	2.3	3.3	2.0	2.9

The Germans expect that after a large sodium/water reaction (i.e., the reaction results in rupturing the rupture disks), they will requalify the steam generator shell but not the tube bundle. The German large leak design basis is that the accident starts with the double ended rupture of one steam generator tube. The basis does not include any secondary ruptures. This basis is strongly supported with their test evidence. Also, they feel that their relief system design is conservative because of the location of rupture disks at both ends of the modules. They stated the double rupture disk location lowers the steam generator peak pressure by a factor of two and greatly reduces the possibility of having secondary tube failures (due to the lower local pressure and the shorter flame front residence time). The German sodium water reaction relief system design is based upon:

- a. Short relief lines
- b. Rupture disks located at each end of the steam generator
- c. Relief piping diameter is sized for two phase, two component flow
- d. The rupture disk design is such that the disk breaks during the initial pressure pulse.

The Germans suggested that the USA may have trouble igniting the hydrogen leaving the CRBR exhaust stack due to the high velocity of the gas stream. If the ignition occurs late, there may be an explosion, and then we would have to show that the explosion pressure is not great. Their experience is that hot sodium droplets in the gas cause spontaneous ignition rather than an explosion; thus an igniter device is unnecessary.

The German sodium water reaction products storage tank (it includes a cyclone separator and storage volume) and exhaust stack is designed for 8 atmosphere pressure and the high temperature of the reaction products. The relief system is initially filled with an inert gas but after the reaction event they do not purge the system with inert gas.

C. Apeldoorn/Stork, October 18, 1974

At the TNO laboratory in Apeldoorn, Netherlands, experiments in the SWISH heat transfer loop with the single tube model of the SNR straight tube steam generator module were nearing completion at the time of the visit. Preparations were underway for the initiation of tests with a single SNR-type helical tube model within about three weeks. Characteristics of the SWISH loop are as listed in the table below:

TABLE II

TNO SWISH LOOP

	<u>Sodium Side</u>	<u>Water Side</u>
Material	SI 316	AISI 316
Maximum temperature	600°C	500°C
Maximum pressure	5 kgf/cm ²	300 kgf/cm ²
EM pump	20 m ³ hour	5 m ³ /hour
Piping/44 mm heater	750 kw	water makeup oxygen content >15 ppb conductivity >0.5 µS/cm

FIG. 6 SCHEMATIC OF SWISH LOOP

Pertinent details of the test unit instrumentation are provided in WASH-1219.

The primary objective in the single model testing is to develop pressure drop and heat transfer information. The loop incorporates a transparent window which permits high speed motion pictures of the steam/water mixture to be taken at the test section outlet. Experiments relative to water side behavior are done elsewhere. The test model and the water system are of stainless steel 316. Water quality in the SWISH loop is similar to that in the 50 Mwt facility at Hengelo, with the exception that the pH of feedwater in SWISH is maintained within the range of 10-11 while that in the Hengelo plant is from 8.6 to 8.9. Neither fouling nor erosion have been observed on the water side. Water side mass flow densities have been as high as 2×10^6 lb/hr sq ft. Similarly, in the SNR steam generator, models tested at Hengelo for 3000 hours, no erosion or fouling was observed under the same mass flow conditions.

It was indicated that studies of water side problems such as fouling and corrosion would probably begin in 1975 after completion of tests on the helical tube model. Little information was provided relative the results of the heat transfer tests; however, it was learned that two anomalies had been observed during the program. At one point, the heat transfer coefficient in the film boiling region increased from its previous values by a factor of 4 for a period of about one week, after which it returned to the original value. At another point during the testing, it was found that the axial position of the DNB point in the tube changed along the length for a short period of time while all other operating conditions remained the same. This occurred with the flow rate at about 50% of the design value.

The two isothermal corrosion loops which are constructed of stainless steel type 316, continue to circulate and have now accumulated six years of operation. The chemistry loop is being employed for further parametric testing of nickel diffusion tube hydrogen detectors. A vanadium corrosion standard is being used for oxygen determination. Mass transfer testing is continuing in the isothermal loop at temperatures ranging between 650°C and 400°C. There has been evidence of deposition in cold sections of the loop; however, there was no indication that this process might lead to fouling that would be of significance to heat transfer efficiency. The thermal shock loop and the pump test loop at Apeldoorn were inactive. The construction of a molybdenum loop was nearing completion. This loop is to be used in "pure" corrosion studies.

During the visit to the Stork facilities, three anode type x-ray units used for examination of tube to tubesheet welds were observed. These units had been developed and supplied by the TNO Institute at Delft. It was reported that these units provided good radioagraphic sensitivity; however, penetrameters had not been used during the examinations. The tube length is such that the source can be placed within the tube bore at a distance of 300 millimeters from the front face of the tubesheet. The tube diameter is 13.5 millimeters and the tungsten target is 0.4 millimeters in diameter.

D. Risley, October 21, 1974

Design Criteria

In the discussion the British stated and evidenced conviction in their belief in the following general design criteria, particularly for application to CFR:

1. The single-wall tube design approach should be continued, and the UK plans involve no serious consideration of double-wall nor protected tube approaches.
2. Removable tube bundles, as used on PFR (bolted flange), should be continued as a prime design requirement -- in order to obviate the necessity of cutting the sodium pipes in the event replacement or extensive out-of-loop repairs become necessary and this to reduce plant down time.
3. "Straight-tubes" (rather than coiled - tubes or serpentine tubes, etc.) are strongly preferred, but with provision for both tube-to-shell and tube-to-tube differential thermal expansion, and definite capability for removal of the tube bundle from the shell, (for example, the UK expressed the opinion that the hockey-stick is a straight-tube and, except for lack of bundle removability, the UK concur it is a good choice for CRBRP).
4. The UK has not yet decided for CFR as to whether to depart from the PFR concept of recirculation with sufficiently high recirculation to avoid DNB. Apparently, a version of the once-through concept which the UK is very seriously considering is to exit from the evaporator (separate evaporators and superheaters) with about 90% steam quality, separate the steam and liquid in a drum, and drain the approximately 10% liquid back to the condensate collection for full-flow demineralizer treatment with the feedwater prior to reheating and return direct to the evaporation. For this system a vertical steam drum is being considered.
5. Tentatively, for the system described in item 4, flow orifices will be used at the inlet of each evaporator tube, sized to give a pressure drop in the orifice approximately

equal to the pressure drop in the remainder of the unit - to ensure flow stability.

6. The UK feels very strongly that isolation valves should be used on the secondary sodium loops, as was done for PFR, in order to provide a means for avoiding or minimizing contamination of the IHX in the event of a large Na/H₂O reaction, to facilitate recovery operation (clean-up and repair) and to enable operation to continue while one loop is shutdown for steam generator repairs. (Note: it was stated that the isolation valves on PFR work satisfactorily, and they are capable of being closed in 10 seconds.) Secondary system valves are also used to control natural convection flow in the secondary circuit to acceptable values. Flow rates as high as 17% of full flow occur.
7. Despite usage of the feature on the PFR, the UK is presently undecided on whether or not to use a cover gas in the units for CFR. In their view, the cover gas is a satisfactory means of protecting the tube sheet from transients, and detectors in the cover gas space provide a rapid means of detecting leaks in the tube-to-tube sheet joints; but they are concerned about reaction product build-up in the cover gas area in the event of a small-leak. Additionally, the strong German opinion based on calculations and tests (INTERATOM), that the cover gas tends to cause a pressure magnification effect during the initial portion of a large-leak event, has contributed to the UK uncertainty on the cover gas question.
8. A heat flux limit of 230,000 BTU/hr-ft² (peak heat flux) has been specified by the CEBG to avoid accelerated corrosion in the evaporator; but owing to the lack of rational consideration of the effects water chemistry and boiling heat transfer have on such a limit, this specific value is not yet concurred in the TNPG nor AERE (apparently)--although the value is said to be derived from survey of fossil-fuel steam generator failures in the UK.(23)
9. The optimum degree of modularization is of concern especially in view of the current small leak experience with the PFR #2 evaporator.
10. In general, the work of preliminary engineering, etc., for CFR is to proceed consistent with intended start of construction in 1977.

Safety Considerations

1. From the discussion, (18) and their operative actions for the existing small-leak in the PFR Evaporator #2 the UK believe that continued operation with a small-leak is dangerous,

and they advocate rapid--even automatic--shutdown upon definite detection that the leak exists.

2. Somewhat contrary to (1), the continued plugging and re-plugging of the leak in the PFR unit associated with repeated attempts to shutdown for leak location and repair (leak plugged, making re-starting necessary in order to open the leak) has resulted in some shift of UK attitude toward the position that operation should be continued until the leak grows to the minimum size necessary for it to remain open (and then capable of being located for tube plugging purposes), after which shutdown for recovery operations would be done.
3. The UK believes that an appropriate design-basis initiating event for the large-leak incident is the equivalent of instantaneous double-ended guillotine fracture of a single-tube. However, it is noteworthy that the UK is not certain on this, and in fact they (a) plan to do leak propagation tests of situations originating from small-leaks, to verify or correct the foregoing; and (b) recognize that subsequent gross tube failures after the initiating event may also occur.
4. Pursuant to 3-b, the design of the PFR is such that after the full-guillotine rupture occurs, six additional guillotine ruptures (or their flow equivalent), sequenced one at a time about 1-2 seconds apart, can also be accommodated safely. It was said that this specific capability is a requirement handed down by the CEGB.
5. Upon occurrence of a large-leak incident, for PFR, and apparently for CFR too, the rupture discs burst, the sodium isolation valves close (in ~ 10 seconds) to prevent spread of reaction products, the steam/water isolation valves close, the affected unit steam/water is blown down (in 30 seconds), the steam/water side is charged with N_2 to prevent sodium entering into the steam/water side, and simultaneously the sodium in the affected unit is rapidly dumped.
6. In contrast to the present approach in the US, the UK does not have an igniter to ensure burning of the H_2 at the relief stack discharge, and does not consider this necessary. Instead, there is used a flapper valve which re-closes in order to economize on the inert gas purge from the system.
7. The UK design-basis calculations of the large-leak event is done with a computer code called FLOOD, which is a quasi-steady-state energy model, employing lumped parameters and one-dimensional, incompressible flow. It is noteworthy that this code does not calculate the initial pressure pulse in

early milli-second range, and in fact only calculates the pressure associated with inertia of acceleration of the fluids and friction/area change pressure forces. The code is rudimentary compared to the US TRANSWRAP II, but apparently the UK uses it for licensing calculations; and in fact stated their opinion that the code is adequate and need not be developed further.

This view is indicated to be based on the assumption that the cover gas acts to mitigate the initial pressure pulses, and that the energy content of the pulse is too small to jeopardize the IHX. What becomes of this position if a cover gas is not used in CFR was not made clear.

Material Selection

The choice of materials for CFR has not yet been made; however, the considerations in process reflect the British view on decarburization, which is that it occurs at a higher rate in unstabilized 2 1/4 Cr-1 Mo than the US is assuming. Because of this, reflecting incomplete acceptance by the British of the US position that decarburization and strength loss in normal 2 1/4 CR-1 Mo is small up to 500°C, the UK is tending to retain the same materials choice for CFR as was used for PFR--Nb-stabilized 2 1/4 Cr-1 Mo for the evaporator and austenitic stainless steel for the superheater. Additionally, as an alternative, the UK is considering 9 Cr-Mo for the superheater. A decarburization test of 9 Cr-Mo has been in operation for about 1 1/2 years at 600-650°C for the purpose of verifying that its carburization rate is small.

Sodium/Water Reaction Tests

The British stated that for the sodium/water reaction testing they have decided to model the "steam generator system" rather than merely test models of the steam generator, including in particular the situation of progression from small-leaks to intermediate-leaks to large-leaks (i.e., leak propagation).

The Super-NOAH rig is being prepared for the fourth test shot that will have been run in this rig since its construction. This test shot (test #4) is scheduled to be done in December '74. The test element for test #4 is the usual Super-NOAH setup, which includes the 6 adjacent tubes surrounding the leak-tube, which are pressurized with inert gas, and the remainder of the tubes in the test bundle unpressurized. The test will be done at 450°C, and the water injection rate is planned to be 45 kg/sec for an interval of 5 seconds.

Subsequent tests in Super-NOAH, beginning with test #5, will concentrate on leaks in the intermediate size range, to obtain information on leak propagation. Specifically, for test #5 a leak of about 0.5 grams/sec using a nominal hole size of 2-3 mm

ID is planned, and as a step toward representing the "steam generator situation" twelve of the tubes in the vicinity of the initiating leak tube will be pressurized with steam/water.

Fabrication and Inspection

The British are actively developing the eddy-current method for tubing inspection, particularly for in-service inspection from the bore-side. They have overcome the problems of magnetic saturation of ferritic material, and the claim was that they can detect a flaw as small as 0.002 inch from the bore-side. It was noted that the search coil and energizing coil should be very close together because the permeability of the tube wall is not constant along the tube length. The miniturization of the eddy current sensor that has been achieved enables successful inspection of 13/16-inch ID tubes from the bore-side, but limited to examination of the inner 0.020-inch radial zone of the tube wall.

For in-service tube inspection (from the bore-side), the British advocate and are actively developing the ultrasonic method for measurement of defects on the tube OD. At the present stage of development of the ultrasonic probe, the British can measure wall thinning type defects (not cracks) on the external tube surface down to 0.002-inch wall thickness variation using the detector inserted in the tube bore. This has been done with 13/16-inch ID tubes, and they are confident that equivalent success can be achieved with 0.5-inch ID tubes, and speculates that development will be successful for the device applied in tubes down to 0.4-inch ID.

Status of PFR

The PFR is currently operating at 10 mw. Loop #3 is steaming. Sodium is circulating in two loops. The steam generator in #2 loop has a small leak and is being exercised to bring the leak to a size which should be locatable. The baffles in the #2 and #3 steam generators have been modified to alleviate the gas entrainment problem. Although the #1 unit has not been modified it is planned to bring the #1 circuit into steaming operation while the #2 unit is being repaired. This requires operating the #1 unit with a flooded tubesheet. The #2 steam generator unit was commissioned during September, 1974. Prior to commissioning, a leak check of the unit indicated the possibility of a very small leak. During a leak test using 500 psi of H₂ and Argonne on the water side, the H₂ content in the shell side rose from 5 to 160 ppm. Subsequently a second test was performed with sodium on the shell side at 400°C. After a few days of no activity the H₂ content of the cover gas rose from 5 to 30 ppm and then gradually reduced to normal background level. Soon after commissioning unit #2 the H₂ detector in the gas space indicated a leak of 10⁻⁵ g/sec, which subsequently disappeared. The unit was operated for a week or so with no further indication

of a leak. Suddenly, the leak opened. After 90 minutes of operation the water in the circuit was drained manually. At this time, the leak size was estimated at 2×10^{-3} g/sec. The unit was again started. After a few hours at pressure the leak opened at an estimated value of 8×10^{-3} g/sec. The unit was operated for 90 minutes and shutdown. At this time, it was believed that the leak could be in the gas space and there was a worry about possible corrosion of the tubes in that area. An attempt was made to run with a flooded tubesheet. Immediately the in-sodium detectors indicated a leak of about 10^{-1} g/sec. and the sodium level was forced down. It is now planned to take the unit out of service and search for the leak.

Proposed 12-MWt Test Facility

A proposal exists for a 12-MWt test facility to perform heat transfer, DNB and corrosion tests. Ten megawatts of the rig would be utilized to test 7 full-length tubes for the CFR evaporator. A 2-Mwt facility operating in parallel to the 1-Mwt unit would be used to study DNB and related corrosion effects. Approval of this facility by the end of 1974 is being sought, in which case the loop would be in operation in 1978.

The British believe that a small test loop (12 MWt) is more useful, timely, and economically justifiable than a large test facility of the Hengelo or Renardieres type. The British opinion is that for large model testing the benefits to be gained by going much larger than 10 MWe, such as to 50 MWt models, is marginal and does not justify the increased cost.

E. Les Renardieres, October 22, 1974

The Renardiere 50 MWt test loop (CVGS) presently has a Stein Industrie straight tube steam generator installed. This is a two module steam generator - one evaporator and one superheater module. This is the third unit to be installed, the first being the Phenix modules, the second being a Babcock-Atlantique helical coil monotube module. Tests on the present Stein design are to be completed early 1975 in anticipation of a choice between the helical coil and straight tube designs by June 1975 for application to Super Phenix.

The facility employs a single sodium circuit. The loop includes one sodium heater, two sodium pumps in parallel with 100% capacity each, an air blast sodium cooler which is in parallel with the steam generators and associated auxiliary sodium systems.

The water circuit was presented in detail. The steam is passed through depressurization and desuperheating stages before entering the condenser. Two feedwater heaters are supplied steam from the two depressurization stages.

Three steam generator models have been installed in the CGVS. Initially, three Phenix type modules were installed and their parallel operations was studied. These tests started in 1970 and were completed in 1972. Dismantling was initiated in January 1973 after having accumulated 7000 hours of steaming on the units. Next, a Babcock-Atlantique helical coil design was installed in mid 1973 and tested with emphasis on performance parameters over the load range including stability. Dismantling of this test configuration was conducted in early 1974 after 100 hours of steaming. Finally a Stein-Industrie straight tube model was installed and is in the early test stage at this time (October 1974). It is planned to complete the same test series as was performed on the helical coil design and to accumulate about 1000 hours of steaming by spring of 1975.

The test series conducted on the helical coil and straight tube designs are both prototypical in tube dimension of the corresponding unit for application to Super Phenix (1200 MWe). The decision to use one or the other in Super Phenix is expected in mid-75 after the current test series is completed.

After selection of one design that unit will be installed (or retained) in the CGVS for extensive testing. These tests will provide thermal shock and endurance data. It is planned to accumulate 3000 to 5000 hours of testing and an objective to achieve up to 10,000 hours of additional steaming exposure on the unit.

The preliminary tests upon which will be completed before a choice of design for Super Phenix include:

1. Full power thermal performance (per tube)
2. Water and sodium side flow resistances
3. Stability of operation over the expected load range and
4. Leak detection characteristics

Experience indicates that these tests require about 1000 hours of steaming and about four months elapsed time.

In discussions concerning the thermal shock testing to be conducted on the selected design, the approach was identified as one of estimating representative transients from the plant evaluations and to simulate these to the best ability of the loop. Supplemental tests are conducted in smaller test facilities elsewhere to obtain detail structural damage assessments. Hence, the CGVS tests are intended primarily to validate the analytical model used in calculating the plant

transients. Quantitative structural damage data cannot practically be obtained in the test loop with such a complex model.

The facility is designed to provide temperature shocks of $+5^{\circ}\text{C}/\text{sec}$ for 10 secs, or $-15^{\circ}\text{C}/\text{sec}$ for 10 secs on the sodium side and $-1^{\circ}\text{C}/\text{sec}$ for 30 secs on the water side. Also sodium or water flow stoppages can be imposed.

The philosophy of model sizing and testing was pursued. In the historical development of facilities original motivation was to establish a new test facility which could drive each tube at its rated conditions. This led to initial loops of about 5 MWt capacity. Fuel costs considerations motivated the smallest size consistent with operating a few tubes in parallel. The CGVS facility size of 50 MWt was strongly influenced by the need to operate three Phenix size modules in parallel (15 MWt each). Subsequent models have been sized consistent with the CGVS capability with strict adherence to providing individual tube operation at rated conditions. Hence the number of tubes becomes the variable in sizing the model. This approach is logical and understandable.

The control room and test instrumentation rooms are adjacent but separate. In this way the operation is more prototypical of a plant. Five teams of four men each are on the operating staff. An on-line test data system includes a computer for compiling and processing data results. The computer can produce 30 to 50 lines of data per second and can follow transients resulting from events such as water side isolation and blowdown, feed water stoppage or sodium flow stoppage.

The Renardiére experience with rupture discs has been excellent. No failures have occurred. Two discs are used in series at each location to provide for detection of a leak in the innermost disc and to facilitate shutdown and repair without loss of sodium. No leaks have occurred as of this writing.

The valve operating experience at Renardiére indicates more difficulties with steam and water valves than with sodium valves. However, as a result of difficulties with frozen seal sodium valves some redesign of the freeze clearance dimension has occurred. After this modification the experience has been judged to be very acceptable to the operating staff. No special periodic maintenance of either sodium or water-steam valves is being practiced.

The CGVS employs a sodium mixing tee where the parallel streams from the air cooler and steam generator are joined. The basic design of this device is shown in item 11 of Appendix C. A failure of the device due to thermal stress cycling occurred

as a circumferential crack in the shell at the entrance of the hot sodium stream which by-passes the steam generator modules.

Upon disassembly of the device some fatigue cracks were observed locally near the tube support region down stream of the hot sodium entrance. A redesign of the hot sodium inlet area was implemented. No other failures have been observed to date.

In a summary discussion the French experience indicates that the main benefit from model testing lies in the confirmation of the structural integrity of the design - especially its fabrication. The thermal-hydraulic performance is believed to be well enough understood to permit unit sizing with acceptable risk. Structural stability and vibration on shell side are achievable from water flow tests conducted elsewhere. Integrity of tube guides are obtainable in Na-H₂O reaction tests elsewhere. The judgment was also expressed that full size prototyping for fabricability is not necessary although some sensitive parts are mocked up extensively; e.g., the tube sheet and tube to tube sheet welds will be prototyped at the full tube sheet size although not with 100% of the tubes.

Leak detection tests have been carried out for each steam generator installed in the CGVS. These test series involve injection of hydrogen or water/steam into selected locations within the module and observing the response of leak detectors. Experience has been that leaks are detected in all detectors, both in sodium or gas space. (The Super Phenix models employ a gas space). Acoustic detection has been tested and the results indicate:

1. Acoustic detectors in sodium are less sensitive than the H₂ detectors by a factor of 10 to 20.
2. Acoustic detectors cannot detect a leak into the gas space. There was some indication that large water side flows masked the acoustic signals, although this was not universally stated. Work on acoustic detection is continuing.

The transient capability which was a part of the basis for design of the loop includes 250 cycles of a cold shock transient at 15 C/sec. over a 150 C range and 250 cycles of a hot shock transient at 5 C/sec. for a 50 C range. The cold shock is provided by by-pass systems in which part of the sodium source is changed from a hot stream to a cold stream. Transients can also be imposed from the water side, also using by-pass systems. Capability includes 1 C/sec. over a 30 C range while maintaining sodium flow.

Details of the Les Renardieres sodium-water reaction relief system were discussed. The reaction tank is fabricated from stainless steel and is designed to take reaction products at

about 600 C. The general design of the tank is shown in the following sketch:

The outlet passes out of the building to the atmosphere. The outlet pipe is covered at the exit end by a plastic rupture disk (low pressure) and a cover falls into place after an incident to allow filling the system with nitrogen. No method is provided for ignition of hydrogen.

The design basis for the Les Renardieres relief system was the double ended rupture of a single 30 mm tube.

The French station personnel questioned the CRBRP relief design. They were concerned that the long pipe from the reaction product tank to the separator would result in excessive pressure buildup in the tank.

A new series of large leak tests will be initiated at Grand Quevilly in support of the Super-Phenix project. This series of 6 or 7 tests will be started about mid-1975. The test section will consist of a tube bundle with approximately 600 tubes (about 1.5 m shell dia.) representing the evaporator.

Within the bundle, 19 tubes, including the initial rupturable tube, will be filled with steam/water at pressure. The remaining tubes will be pressurized with nitrogen. The sodium temperature within the bundle will be 370 C.

F. Stein Industries, October 23, 1974

Stein Industries has designed and fabricated a significant number of components for the French LMFBR program. Their experience to date includes the IHX, steam generators, and sodium storage tank for Rapsodie, several test models for the Phenix steam generator development program and, subsequently, the secondary sodium loop for Phenix, including the IHX, steam generators, and the sodium purification system. In addition, they have designed and provided components for portions of the 50 MWt sodium test loop at Les Renardieres.

Stein is presently engaged in studies and testing work on components for the projected Super Phenix project. Included is the construction of the model sine wave steam generator which is installed at Les Renardieres and will be shortly tested for a minimum of 1000 hours.

The engineering offices are located near Paris (the site of our visit) but the manufacturing facilities are located in northern France near the Belgium border, about 150 miles away. Additional information obtained at Stein Industries is found in the following pages under the material listed as Super PHENIX (see pages 58-61).

G. Phenix, October 24, 1974

Phenix was operating at 100% power during the visit on October 24. Refueling of 1/6 of the core had just been completed with the fuel having accumulated to date an exposure of 10,000 Mw days per ton. Four Super Phenix type fuel assemblies were installed during the refueling operation. The 250 Mw of Phenix represents 1% of the total electrical generating capacity in France, and approximately 9% of the nuclear capacity. The plant reached full power operation in March 1974, and it has operated on an essentially continuous basis since June 1974. The performance of the steam generators in Phenix has been excellent and has been consistent with expectations based on the earlier tests at Les Renardieres and Grand Quevilly. A total of 630 million kw hours of power production has been recorded.

In September, a sodium leak developed at the butterfly valve which controls the split in sodium flow between the reheater and superheater banks. Leakage occurred through a seal weld located at the bottom of a housing which encloses the bearing for the shaft of the butterfly disc. Leakage also occurred at

the butt weld where the valve interfaces with the adjoining piping system. It is not clear whether the leakage at the two sites was independent or whether the leakage at one site led to that at the second. The French indicated that the butt weld had required repair at the time of construction. One of the two leak detectors in the vicinity of the leak had signaled the possibility of leakage, but since the second detector did not provide confirmation, normal operations were continued for an additional two weeks. At the end of this time, the appearance of sodium smoke confirmed the existence of a leak. The fix consisted of replacing the valve. This required a period of two weeks and was accomplished while the other two loops remained in operation. Sodium was drained from the secondary loop and isolation valves in the hot and cold leg were closed while the repair work was being carried out. After the new valve had been installed, the welds were examined by radiography and helium leak test, and the loop was returned to service.

Separate reaction products tanks and relief system manifolds are provided for each of the three steam generator banks. The reaction products tanks are approximately 4 meters in diameter by 7 meters tall. Rupture discs are located in the inlet and outlet headers of the evaporators and reheater sections and discharge to the main relief system header which runs vertically. The main header is of 600 mm diameter and it dumps tangentially into the reaction products tank, which produces centrifugal motion of the reaction product stream. Several separating devices are provided at the points where the laterals tie into the header for the purpose of providing preliminary separation of reaction product components prior to entry into the reaction product tanks. The reaction product tanks are sized to handle the total sodium inventory of the secondary loop, and incorporate separators at the point of entry of the relief header. The hydrogen stack is mounted top-center on the tank and is of 800 mm diameter.

During the initial period of Phenix operational tests, considerable time was devoted to the calibration of the hydrogen detectors by means of hydrogen injections. No problems have been experienced with spurious signals since that time. At the time of the visit, the background hydrogen levels observed in the plant were as follows:

<u>Location</u>	<u>Hydrogen Level</u>
Superheater exit	70 ppb
Evaporator exit	74 ppb
Reheater exit	110 ppb

When the indicated hydrogen level indicated by the detector reaches twice the normal background level, an alarm will sound. If the rate of increase in the indicated hydrogen level is sufficiently high (value not given), the operator will initiate a rapid shutdown of the loop. At a lower rate of rise in hydrogen level (value not given), steps would be taken to locate the leaky steam generator unit by successively sampling hydrogen in exit streams from various modules. At a still lower level, the operator will continue to observe the hydrogen level until there is some positive indication that a leak exists.

The Phenix plant is equipped with a condenser having brass tubes, and feedwater heater tubing is of carbon steel. The normal feedwater chemistry maintained is as indicated in Table III.

Table III

	<u>Theoretical Level</u>	<u>Practice</u>	<u>Remarks (See Notes)</u>
Fe	>10 ppb	5 ppb	(1)
O ₂	>10 ppb	2	(1)
H ₂	--	2	(1)
SiO ₂	>10 ppb	>5 ppb	(1)
Cu	>5 ppb	> ppb	(2)
Cond	>0.1 μ S/cm	0.07 μ S/cm	(1)
pH	9.2/9.4	9.2/9.4	
N ₂ H ₄	--	10/20 ppb	

(1) Continuous signal sent to computer

(2) Measured daily

Water samples for laboratory analysis are obtained at the feedwater inlet, and conductivity and hydrogen levels are sampled at the exits of the evaporator, superheater, and reheater sections. Hydrozene and ammonia are added for corrosion and pH control. Samples obtained for turbidity measurements have generally been found to consist of 19% FeO₂, 10% Cu, with the balance being manganese.

When the plant is shut down for refueling operations, water/steam will be dumped from the units when the decay heat power level is sufficiently low that heat removal can be accomplished

by air cooling of the modules. During this time, the tube side of the units is back filled with nitrogen.

PHENIX Current Status

On October 24, 1974, the day of the U.S. Team visit the reactor was operating very smoothly at full power. The key plant parameters were as follows:

Reactor Power	562 MW
Electrical Power	250 MWe (gross)
Primary Na Outlet Temp.	560°C (1040°F)
Primary Na Inlet Temp.	400°C (752°F)
Steam Temp. at Throttle	512°C (953°F)
Steam Pressure at Throttle	163 bar (2360 psi)
Primary Pump Speed	810 rpm
Secondary Pump Speed	800 rpm

Those who follow PHENIX activities closely will recognize how remarkably closely these actual operating conditions match the design values published 3-4 years ago.

As is customary for PHENIX, the plant is operating in a base-load manner, with reactor power held constant, sodium pump speeds manually adjusted to maintain coolant outlet temperature at 560°C, and the steam system on automatic boiler pressure control. By October 24 the plant had produced 649×10^9 kwh of gross electrical energy and the operations people said they expected to reach a total of 1000×10^9 kwh by the end of December. It will be recalled that PHENIX first achieved criticality on August 31, 1973, delivered its first electricity to the grid on December 13, 1973, and reached full rated power, only seven months ago, on March 12, 1974 and has operated on an essentially continuous basis since May 1974. The 250 MWe PHENIX represents 1% of the total electrical generating capacity in France and 8.2% of the nuclear capacity.

PHENIX Fuel and Core

So far, the peak burnup on PHENIX driver fuel is still a modest 18,000 MWd/tonne oxide.

Fuel changing operations have been performed three times since last March. The full particulars of these changes were not made clear. The French hosts would say in only general terms that about three assemblies were changed in May, about two in late July and another three in late October.

None of the 48 original fuel assemblies which had contained only UO_2 fuel, enriched to 27% U-235, have yet been removed from the reactor. Also, there have been no changes among the twelve higher enrichment $(\text{U-25Pu})\text{O}_2$ mixed-oxide subassemblies, which are located in pairs on the flats of the sixth hexagonal ring.

There are now only three diluent assemblies remaining in the reactor all in the third hexagonal ring. The operations people said there had originally been six diluent assemblies in the core, but most authoritative sources report that there were nine.

Five low enrichment $(\text{U-18Pu})\text{O}_2$ mixed-oxide fuel assemblies are presently cooling in some in-vessel fuel storage positions. These are CPa-0012, CPa-0003, CPa-0006, CPa-0008, and CPa-0011.

Sodium Pump Operation

Besides the record breaking achievements of the reactor, the PHENIX sodium pumps are achieving new records in their own right. Although slightly smaller in rating (16,650 gpm) than the PFR primary pumps (21,120 gpm) and the SNR prototypes (21,900 gpm) they have surpassed these others in total operating time. Table IV shows the operating hours of the three PHENIX primary pumps and the spare as of the end of September 1974. Every primary pump has been removed at least once, for modification of the anti-reverse-flow valve, so that the pumps are not necessarily in their original circuits. We were told that P_1 , P_2 , and P_4 were currently in service. Table IV also shows the hours of operation of the secondary pumps. None of the secondary pumps has been removed from its circuit and there is no spare available, although there is one spare primary pump motor which can be used if necessary to drive a secondary pump.

Table IV

PHENIX Sodium Pump Operation

Cumulative Hours of Operation as of September 1974

P_1	P_2	P_3	P_4 (Orig. Spare)
8,075	8,112	5,206	4,877
S_1	S_2	S_3	P_1 = primary loop pump # 1
10,729	10,500	11,097	S_1 = secondary loop pump #1

By way of comparison, the two 3,170 gpm sodium pumps at Les Renardieres have operated a total of 22,000 hours, together. Each of these is rated for 100% capacity and therefore only one is in service at any time.

PHENIX Staff

There are now about 215 people at PHENIX including 20 engineers. These numbers are expected to decrease to an equilibrium level of 150 people including 10 engineers by early next year.

About 80% of the people work for CEA and the rest for EdF. The director is M. Conte of CEA and his deputy is M. Godde of EdF. Presently there are five departments: Operations, Maintenance, Fuel Handling, Physics and Administration. By next year the Physics and Fuel Handling activities will be folded into the Operations and Maintenance departments. There are four shifts of eight men each. Except for fuel handling, which is done during shutdown, all operations are conducted from the one central control room. The reactor hall and steam generator hall are accessible during full power operation, not only to staff but to visitors.

PHENIX Neutron Source

The Sb-Be neutron source still remains in the reactor, in core position 13/20 where it has been since it was moved from the central position 20/20 during the initial core loading. M. Leduc commented that he believes a neutron source will be utilized in SUPER-PHENIX even though in that reactor, all the initial fuel assemblies will be fueled with mixed (U-Pu) oxide.

Limited discussion was centered upon the Phenix steam generator design and fabrication techniques.

It was revealed that the Phenix steam generator was fabricated at Stein Industrie and then shipped by truck to the site, where the modules were assembled to the headers. Since the evaporator modules are fabricated from 2 1/4 Cr - 1 Mo, a ferritic to austenitic weld exists in the vicinity of the headers of the PHENIX steam generators. During fabrication of the modules, an inconel transition is welded to the ferritic steel and heat treated. The field weld of the module to header therefore requires no preheat or heat treatment. (Note that this is not possible in the superheater under current US high temperature rules since Inconel is not yet justified for high temperature service).

During shipment of the PHENIX modules, no protection was given the outside surfaces of the modules. The interior surfaces,

however, were protected by sealing with welded caps and pressurizing with N₂.

Stein indicated that they had limited materials testing capability.

In the PHENIX tests at Les Renardieres, three different materials were used for the evaporators. Included were 2 1/4 Cr - 1 Mo, 316SS (?), and a ferritic alloy designated KM-33, stabilized with Vanadium and Niobium.

In response to questions concerning velocity limits on the steam side, Stein personnel related that velocity in the bends of the PHENIX evaporator was approximately 10 - 12 meters/sec. No problems were experienced. In gas cooled experiments even higher velocities have been experienced at 30 bars. They mentioned that care must be taken to remove foreign particles from the components in order to avoid possible erosion where flow direction changes.

When asked about double wall steam generators, they cited experience with a test model in Grand Quevilly in 1964. They are concerned about problems due to complicated fabrication techniques and difficulty in inspection.

Stein Industrie also stated that no spare steam generators have been manufactured for PHENIX.

H. SUPER PHENIX

A considerable amount of discussion centered upon the design and fabrication of the Super PHENIX test models and steam generators.

Design

The Stein Representatives presented the following data on the sine wave steam generators that are installed in Le Renardieres and on those that are planned for Super PHENIX.

Table V

Component:	Les Renardieres Test Model		Super PHENIX Generator	
	Evap.	Sup Htr	Evap.	Sup Htr
Power (MWt)	33.29	11.608	561.0	189.0
Number Tubes	102	37	1405	595
Tube O.D. (mm)	18	21.3	18	21.3
Tube Wall (mm)	3	2.3	3	2.3
*Heat X-Fer Length (m)	17.33	16.36	22.85	19.26
Shell O.D. (mm)	676	600	2340	2248
**Tube Mat'l	2 1/4 Cr - 1 Mo (.07 - .1%C)	316SS	2 1/4 Cr - 1 Mo (.07 - .1%C)	316SS

*Add 2.5 m to active length for TL length

**Confirmed by Mr. Salon

Note: Expansion wave length 2.5 m.

It was noted that no tube supports are located in the bend region with the sine wave design.

Stein utilized various mathematical models to predict shell side flow distribution in their steam generators, including computer models employing resistance networks. They stated, however, that they did not have enough confidence in these models to eliminate water flow testing. In fact they use the tests to tune the computer representations.

Materials

The tubesheets for the sine wave components are made to ultra-clean practice (VAR or equivalent). The tubing, however, is not necessarily of a special material. They rely instead upon a very tight ultrasonic standard for the tubes (.2 mm notch depth), and other NDE before, during and after fabrication.

Tubes are finished at the vendor's plant by cold drawing and are shipped with end caps, wrapped in plastic and crated. (Stein bends the tubing at their plant.)

Tube to Tubesheet Welding

In welding the tubes to the tubesheet in the sine wave steam generator, Stein Industrie uses a weld prep machined into the tubesheet face, similar to the CRBRP design. Stein Industrie also investigated welds of the "corner" configuration, similar to SNR, but selected the "spigot" prep due to the smaller holes required in the tubesheet, simpler RT techniques and the potentially larger problems with maintaining preheat and with post weld heat treatment in the SNR design.

Endurance tests have been made on the selected configuration. The results were termed satisfactory, but the implication received was that some of the fatigue tests resulted in failure at the weld as opposed to within the tubing.

The tubes are welded by a TIG process in the horizontal position without filler metal addition. Some force is applied between the tube and tubesheet during the welding process. Welding samples indicated more reinforcement at the weld location than encountered in the U.S. process.

The ferritic tube to tubesheet welds may or may not be welded using preheat. If preheat is used, it is dropped prior to radiography. Post weld heat treatment is employed using local techniques, but Stein questions its value with thin tube walls.

The tube to tubesheet welds are made in rows with radiography following completion of rows. A thulium source is used rather than X-ray for radiography.

A rather unique NDE method is also employed for testing tube to tubesheet welds. The tubes are pressurized with NH_3 to about 2 1/2 bars and litmus paper on the outside of the tubes is used as a detector. Leaks are specifically located by a color change in the paper. Stein claims that when this method has been used, no subsequent leaks have been found by a He leak test. (They do, however, perform a He leak test on the completed assembly.)

In the course of qualifying the tube to tubesheet for the straight tube test model, Stein performed in excess of 1000 welds. Some trouble was encountered, however, when a modification to the weld was made in the test model.

The tube to tubesheet welds for the full size unit will be qualified on a full scale tubesheet mockup. A large number of these welds will be made.

Other Fabrication Techniques

Other fabrication techniques used at Stein Industrie are largely conventional.

Component parts of steam generators are acid cleaned before assembly.

Shell pieces are assembled and the closing welds are radiographed using double wall techniques.

After completion of assembly, the modules are pressure tested using alcohol as the fluid. This also constitutes the final step in cleaning. The modules are dried using a vacuum and are subsequently given a He leak test. Stein is uncertain of the value of the final He test.

During assembly of LMFBR components, Stein employs a limited access work area with barriers so that transient traffic is limited. These areas are in all other respects open to the shop environment. They do not employ clean rooms for these applications.

Standards

The French do not concentrate their attention on standards to the extent seen in the U.S. program. The approach appears to be more pragmatic, in that new standards are developed only where a need is clearly indicated. When the need for a standard is identified, it is jointly developed by the CEA and the manufacturer who is likely to employ it. This philosophy results in greater acceptance by suppliers and minimum added expenditures.

SUPER PHENIX IHX

A minimal discussion was held on the Super Phenix IHX. The design incorporates 5380 tubes 14 mm O.D. x 12 mm I.D. approximately 6.5 m long. The bundle diameter is about 2.2 m. Water testing of the tube side indicated the need for flow guides inside the head to equalize flow between tubes. The tubes are rolled and seal-welded to the tube-sheets and no expansion bends are employed.

SUPER PHENIX Core

It will be recalled that there has been the strong expectation that the SUPER PHENIX, like PHENIX, would have to be loaded partly with enriched UO_2 , because the available plutonium in France is not sufficient to provide full mixed oxide cores for these early FBR's. This problem now appears to have been solved, and SUPER PHENIX will definitely have an all mixed oxide core. A decision made last month (September 1974) by the three owner utilities provides that the French EdF will provide 51% of the Pu, the Italian ENEL-33% and the German utility RWE-16%, each in proportion to its project participation. One problem with this arrangement is that the fuel will come from a variety of thermal reactors creating somewhat of a "mixed bag" of plutonium isotope ratios.

There is also some likelihood that Italy will produce the fuel pellets and fuel pins for its share of the core, to the same specification as the French pins, of course.

SUPER PHENIX Plant

Turbine Generators

At the time of the London BNES meeting J. Megy of TECHNICATOME reported that most of the major decisions on SUPER PHENIX had already been made, with two notable exceptions, 1) whether there would be one or two turbine-generator sets; and 2) what type of steam generator would be chosen. The first of these decisions was made last month--to have two 600 MWe T-G sets, based on the limited technology on 1200 MWe generators in France, according to Leduc.

The choice of steam generator, discussed elsewhere in this report, is expected to be made in May or June 1975.

Pumps

The SUPER PHENIX pumps will likely be made by the same firm that made the PHENIX pumps, Hispano-Suiza, except that the Italian Fiat may make some contribution to the pump order for the new plant.

Hydrogen Detection

Whereas PHENIX samples the sodium from each steam generator module, analyzes it for hydrogen and then corrects the reading for temperature, by use of computer, SUPER PHENIX will adopt the system used by most other programs in the world. That is, the sodium temperature from the sample streams will be temperature-normalized before the hydrogen membrane.

IV. APPENDICES

A. Appendix A - List of IAEA Papers

Design and Construction of Past and Present Steam Generators

1. O. Hayden
Design and construction of past and present steam generators for UK fast reactors
2. P. A. Lowe
Overview of the U.S. steam generator testing program
3. M. Kawara et. al.
Design and construction of 50 MW steam generator
4. Y. Hoshi et. al.
Design and related R&D works of "MONJU" steam generators
5. J. Essenbaggers
The Neratoom straight tube and helical coil tube steam generator design
6. F. E. Tippetts
Clinch River breeder reactor steam generator system design
7. P. G. Avanzini, et. al.
Design philosophy and criteria for CNEC-NIRA straight tube steam generator
8. P. G. Avanzini et. al.
Technological aspects of the CNEC-NIRA steam generator and its experimental program

Criteria for Sodium-Heated Steam Generators

9. P. Bolt
Aspects affecting steam generator unit integrity considered from the viewpoint of a utility

Safety Aspects

10. M. Hori et. al.
Sodium-water reaction test for "MONJU" steam generator
11. P.W.P.H. Ludwig
Conclusions from the sodium-water reaction experiments with a straight tube bundle steam generator model with respect to the calculation of the design pressure of steam generators

12. K. Sakano et. al.
Large-leak sodium-water reaction analysis for steam generators
13. F. E. Tippets
Sodium-water reaction, relief system design
14. K. Dumm, M. Fortmann
Development of rupture discs for LMFBR steam generators
15. J. Biscarel, A. Lafon, N. Lions
Essais de petites fuites d'eau dans le sodium sur les tubes des generateur de vapeur de Phenix
16. K. Dumm
Small water/steam leaks in sodium-heated steam generators
- Evaluations on the reaction zone and
- Effects on 2 1/4 Cr 1 Mo structural material
17. M. Hanawa et. al.
Metallurgical examination of wastage materials
18. J. A. Smedley
Implications of small water leak reactions of steam generator design
19. W. R. Simmons
Evaluation of leak detectors in SCTI
20. E. Cambillard, J. Lacrois, G. Langlois, M. Viala
Generateurs de vapeur de Phenix -
Mesure de la concentration d'hydrogene du sodium pour la surveillance de l'etancheite des tubes d'eau-vapeur
21. M. Hissink
Leakage detection
22. K. Dumm
Small leak detection requirements for SNR-300 steam generator operation

Operating Conditions

23. P. Bolt
Steam generator unit thermal rating selection to minimize possible water-side corrosion problems
24. W. Ratzel
Operating conditions of steam generators for LMFBR's

Materials

25. C. Tyzack
The selection of fast reactor boiler tube materials
26. C. N. Spalaris
Croloy metallurgy
27. E. D. Gosser, G. Menken
Corrosion allowances for sodium-heated steam generators
- Evaluation and extrapolation of influences to
component life-time

Maintenance and Repair

28. C. N. Spalaris
Non-destructive examination (part of paper 26)
29. R. W. Devlin
Field Repair
30. J. Essenbaggers, C. C. Smit
Maintenance and repair of steam generators
- Cleaning methods
31. J. S. McDonald
Technical considerations relative to removal of sodium
from LMFBR components

Realization of the Criteria for Past and Present Steam Generators

32. M. G. Robin, L. Duchatelle, L. de Nucheze
Performances des generateurs de vapeur de Phenix en
regime permanent
- Comparaison avec les previsions
33. P. J. de Munk
Wall temperature fluctuations
- Realization of the criteria for steam generators

Operating Behavior

34. J. S. McDonald
Summary results from sodium-heated steam generator test
program
35. A. Sano et. al.
Operating experiences with 1 MW steam generator
36. A. Kanamori et. al.
Operating experiences with 50 MW steam generator

37. P.W.P.H. Ludwig
Some results of the 50 MW straight tube steam generator-
tests in the TNO-facility at Hengelo
38. A. Urbani, P. Valentini
Operation and post-operation examination on the 1 MW test
section SDP - 1
39. K. Vinzens
Heat transfer in the boiling zone of the bajonet type
reaction protected integrated steam generator

Water Metal Reactions

40. P. A. Lowe
Water-side chemistry
41. W. Haubold, E. Buscher, W. Jansing, K. Vinzens
Experimental studies on water-side corrosion problems
of sodium-heated steam generators

Steam Generator Faults

42. A. Kanamori et. al.
Leakage experiences with 1 MW steam generator
43. K. Dumm, W. Ratzel
KNK-steam generator leakage
- Evaluations and improvements
44. E. D. Grosser, H. Lorenz
Metallurgical post-examinations of KNK-steam generator
failure
45. C. C. Smit
Steam generator faults

Future Design Aspects

46. P. A. Taylor, O. Hayden
Questions raised in developing a design for commercial
fast reactor steam generator units
47. M. G. Robin
Générateurs de vapeur développés en France pour
Super Phenix
48. P. G. Avanzini, S. de Leo
Straight tube steam generator evolution with CNEN-NIRA
concept
49. W. Jansing, K. Vinzens, W. Ratzel
Future aspects for liquid metal-heated steam generators
50. J. Essenbagggers
Future steam generator development activities of NERATOOM

B. Appendix B - List of Attendees at IAEA

Program Coordinators

H. Mausbeck, K. Forster, A. H. Spano, K. Vinzens

C.S.S.R.

F. Suransky
V. Krizek
F. Dubsek

France

J. M.	Saur	Technicatome, Saclay
H.	Halle	l'Electricite de France
J.	Biscarel	Centre d'Etudes Nucleaires de Cadarache
M.G.	Robin	Centre d'Etudes Nucleaires de Cadarache
E.	Cambillard	Centre d'Etudes Nucleaires de Saclay
A.	Lafon	Division d'Etudes et de Development des Reacteurs, Gif-sur-Yvette
G.	Langlois	Technicatome, Saclay
G.	Cornu	EdF-REL, Group-Rapide, Lyon

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E.	Buscher	Interatom, Bensberg
K.	Dumm	Interatom, Bensberg
E. D.	Grosser	Interatom, Bensberg
W.	Haubold	Interatom, Bensberg
W.	Jansing	Interatom, Bensberg
W.	Ratzel	Interatom, Bensberg
H.	Knuth	Kernforschungszentrum, Karlsruhe

Italy

C.	Salgo	CNEN
P. G.	Avanzini	F. TOSI
V.	Bedogni	F. TOSI
A.	Gerosa	CNEN
P.	Valentini	NIRA
F.	Bottiglioni	CNEN
P.	Casalini	CNEN

Japan

M.	Hori	PNC, Oarai
A.	Kanamuri	PNC, Oarai
Y.	Hoshi	PNC, Tokyo
A.	Sano	Ishikawajima-Harima Heavy Industries
M.	Hanawa	Mitsubishi Atomic Power Industries
K.	Sakano	Kawasaki Heavy Industries
M.	Kawara	Hitachi

Netherlands

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C. C.	Smit	Technical Industrial Organization
A. H.	de Haas van Dorsser	Neratoom
J.	Essenbaggers	Neratoom
P.W.P.H	Ludwig	Neratoom
J.	van den Bergh	Stichting Reactor Centrum Nederland
N.	Beyer	
T.H.J.	Tromp	Neratoom

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C.	Tyzack	UKAEA, Risley
J. A.	Smedley	UKAEA, Dounreay
O.	Hayden	TNPG
F.	Marshall	UKAEA, Culcheth
P. A.	Taylor	Babcock & Wilcox
A.	Chitty	Clarke-Chapman

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A. S.	Sokolov
V. S.	Sroelov

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S. R.	Penfield, Jr.	Combustion Engineering
J. S.	McDonald	Atomics International
W. R.	Simmons	Argonne National Laboratory
F. E.	Tippets	General Electric
R. W.	Devlin	Westinghouse
W. B.	Wolfe	USAEC Representative to INTERATOM
W.	Wolowodiuk	Foster-Wheeler Corporation

C. Appendix C - List of Materials Received

1.	IMPIANTO ISA-1	Brochures by
2.	The Brasemone Area	COMITATO NATIONALE
3.	Satana Test	ENERGIA NUCLEARE
4.	Sarcino Test	(all in Italian)
5.	Interaton Sodium Technology Center Brochure (in English)	
6.	50 MW Sodium Component Test Facility. Brochure by TNO.
(in English)	
7.	Fast Reactor Technology by TNO (in English)	
8.	1964-1974 Dix ans d'experiences au Centre d'essas
Thermiques du C.E.A. a Grand Quevilly (in French)	
9.	Grand Quevilly Thermal Test Facility Brochure by CEA
(in English)	
10.	Rapsodie, Combustion, No. 147, July 1969 (in French)	
11.	CGVS/50 MW Brochure by EdF (in French)	
12.	Stein Industrie, Combustion Review about Phenix, Sept. 1972
(in English)	
13.	Nuclear Activities of Stein Industries, 1972 (in English)	
14.	Stein Industrie Brochure (in French)	
15.	Phenix, Combustion, No. 153, Sept. 1972 (in French)	
16.	Phenix Brochure by CEA/EdF/GAAA, 1971, (in English)	
17.	Phenix 450 Brochure (in English)	